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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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564th Meeting

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WEDNESDAY,

JUNE 8, 2009

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ROCKVILLE, MD

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The Committee convened in Room T-2B3 in the Headquarters of the Nuclear Regulatory Commission, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, at 8:30 a.m., Dr. Mario Bonaca, Chair, presiding.

COMMITTEE MEMBERS PRESENT:

MARIO V. BONACA, Chair

SAID ABDEL-KHALIK, Vice Chair

J. SAM ARMIJO, Member-At-Large

JOHN D. SIEBER

SANJOY BANERJEE

JOHN W. STETKAR

DENNIS C. BLEY

COMMITTEE MEMBERS PRESENT: (CONT.)

DANA A. POWERS

WILLIAM J. SHACK

MICHAEL T. RYAN

OTTO L. MAYNARD

HAROLD B. RAY

CHARLES H. BROWN, JR.

MICHAEL CORRADINI

GEORGE E. APOSTOLAKIS

NRC STAFF PRESENT:

KENT HOWARD

SAMSON LEE

BRIAN HOLIAN

RON BELLAMY

DUC NGUYEN

DAVE WERKHEISER

HANSRAJ ASHAR

BENNETT BRADY

CHING NG

JOHN BURKE

GOUTAM BAGCHI

STU RICHARDS

KAMAL MANOLY

TOM SCARBROUGH

PEI-YING CHEN

NRC STAFF PRESENT: (CONT.)

ANNE KAMMERER

CHRIS HOXIE

RALPH LANDRY

STEVE BAJOREK

JOSEPH STAUDENMEIER

ALSO PRESENT:

JOHN THOMAS

MARK A. MANOLERAS

CLIFFORD I. CUSTER

PETER P. SENA, III

BRIAN MURTAGH

STEVE BUFFINGTON

DENNIS WEAKLAND

BILL ETZEL

DAVE GRABSKI

P.T. KUO

RICHARD STARCK

JIM PARELLO

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P R O C E E D I N G S

8:28 A.M.

CHAIR BONACA: Good morning. The meeting will now come to order.

This is the first day of the 564th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the Committee will consider the following: license renewal application in the Final Safety Evaluation Report for the Beaver Valley Power Station; Draft Final Revision 3 to Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Power Plants"; Applicability of TRACE Code to Evaluate New Light Water Reactor (LWR) Designs; Format and Content of the Biennial Research Report to the Commission on the NRC Safety Research Program; and preparation of ACRS reports.

A portion of the session dealing with applicability of the TRACE code to evaluate new Light Water Reactor designs may be closed to discuss information that is proprietary to General Electric Hitachi or its contractors.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Mr. Sam Duraiswamy is the Designated Federal Official for the initial portion of the meeting.

We have received written comments from Paul Gunter of Beyond Nuclear regarding the Beaver Valley license renewal applications. His comments will be made part of the record of today's meeting.

We have received no requests for time to make oral statements from members of the public regarding today's sessions. Federal and industry personnel will be on the phone bridge line to listen to the discussion regarding Regulatory Guide 1.100 and TRACE Code.

To preclude interruption of the meeting, the phone lines will be placed in a listening mode during the presentations and Committee discussion. A transcript of a portion of the meeting is being kept. It is requested that the speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can be readily heard.

I will begin with some items of current interest. Board members who have not completed a mandatory online training course on information security awareness should complete it during this week. If you need assistance, see Vicky Brown.

Mr. David Bessette, who has been with the

NRC for about 30 years, of which about three years with the ACRS staff, has retired on June 30, 2009. During his tenure on the ACRS he provided technical support to the Committee in its review of several matters including PWR sump performance, applicability of the TRACE Code to the ESBWR design, and power uprate applications. His in-depth knowledge of thermal hydraulic issues, regulatory process, and technical support to the Committee reviewing several complex, technical issues are much appreciated. We wish him good luck in his future endeavors.

We have several new staff members and summer hires. I will present their bios and please hold your applause until I finish reading the bios.

(Laughter.)

MEMBER CORRADINI: Are they going to stand up so we can find them in the room?

CHAIR BONACA: We will ask them to stand up at that point.

New staff members: Ms. Kathy Weaver joined the ACRS staff as a Senior Staff Engineer on June 8, 2009. She has been with the NRC since 1990. Prior to joining the ACRS staff, she worked as a reactor inspector, a resident inspector in Region 4, a senior resident inspector in Region 2, a senior project manager in NRR and mostly simply the technical assistant to the NRR, associate director for Operating Reactor Oversight and Licensing. Ms. Weaver received a Bachelor's degree in Engineering and an Associate degree in Nuclear Technology from Arkansas Technical University. She will be the Cognizant Staff Engineer for the Plant Operations and Fire Protection Subcommittee.

Dr. Weidong Wang joined the ACRS staff as a senior staff engineer on July 6, 2009. He has been with the NRC since 1999. Prior to joining the ACRS staff, Dr. Wang worked at the Office of Research as a Reactor System Engineer. From 1999 to 2006, he managed a number of research projects including PUMA, experimental problems in the TRACE and RELAP code development project. In 2007, he joined NRR and reviewed ESBWR design certification, ESBWR COL, and ABWR COL applications. Technical areas he reviewed include the ESBWR LOCAs, instability, transients and applicability of TRACE for analyzing the ESBWR design. Prior to joining the NRC, Dr. Wang worked at INL, Idaho National Laboratory where his main responsibilities included reactor system code development and code user and support. Dr. Wang graduated from Suzhou University in China with a

Bachelor's degree in Physics in 1983 and received his Ph.D. from the School of Nuclear Engineering of Purdue University in 1997. Dr. Wang will be working with thermal hydraulic issues, PWR performance, EPU applications and other issues as assigned.

Ms. McKoy Moore joined the ACRS staff as a team leader in June 2009. She has been with the NRC since 2007. Prior to joining the ACRS staff, she worked as a recruiting and professional development coordinator for the Office of the General Counsel. Ms. Moore has over ten years of experience in workforce and professional development which includes diversity and professional development, manager for Robins, Kaplan, Miller and Ciresi, assistant director for career and professional development, University of St. Thomas School of Law and staff attorney for workforce development and public benefits, Mid Minnesota Legal Services. She's a recent graduate of the NRC Leadership Potential Program and holds a juris doctorate from the Howard University School of Law and a Bachelor of Arts in Sociology from the University of North Carolina.

Ms. Desiree Davis joined the ACRS staff in June as a management analyst. She holds a B.A. degree in psychology and a B.A. degree in French Language and

Literature from the University of Maryland, College Park. This fall, Desiree will pursue a master's degree at the Georgetown University majoring in Security Studies with a concentration in international security.

And finally, summer interns: Ms. Gabrielle Fuller joined the ACRS staff recently as a summer intern. Gabrielle is currently pursuing a master's degree at the College of New Jersey majoring in interactive multimedia with a minor in women and gender studies.

Mr. Thomas D'Agostino joined the ACRS recently as a summer intern. Thomas is pursuing a B.S. degree in Civil Engineering at Virginia Tech. Subsequent to graduation, he plans to pursue a master's degree in Civil Engineering. He's currently assisting Mike Lee with a paper on seismic safety in nuclear reactors.

And finally, Mr. Patrick Arzabarzin joined us on staff as a summer intern in June. He's pursuing a B.S. degree majoring in political science at Purdue University. He is currently involved in the ACRS conference room renovation project. Subsequent to graduate, Patrick plans to pursue a career in politics or work as an attorney for the Federal Government.

With that I think I listed all of them and all of you welcome aboard.

(Applause.)

Okay, that was quite a number of new arrivals.

(Off the record comments.)

We can move now to the first item on the agenda which is license renewal application and final Safety Evaluation Report for the Beaver Valley Power Station and Dr. Bley will lead us through that presentation.

MEMBER BLEY: Thank you, Mr. Chairman. Beaver Valley Power Station Units 1 and 2 are 3-loop Westinghouse PWRs situated on the Ohio River, a bit down river from Pittsburgh. The current license power rating of each of the units is 2900 megawatt-thermal and gross electrical output of 974, 969 megawatts for Unit 1 and 2 respectively.

First Energy requested renewal of the operating license for 20 years beyond the current license terms which expire in 2016 for Unit 1 and 2027 for Unit 2. One thing I'll mention before we get into the presentation is we had a subcommittee meeting back on February 4th. One of the impressive things to me was they really managed to have minimal exceptions to the GALL. I think they met 92 percent of those.

Coming out of that subcommittee meeting, there were several more RAIs issued and resolved. I think we'll hear about those. And at that meeting, our members raised several concerns, the most important of which us seemed to be the issue of submerged 4kV cables for the relevant servicewater pumps and today we're going to hear how that's been resolved.

Beaver Valley 1 containment liner corrosion, which a number of the members expressed real concern about, especially the issues of how convinced can we be that no water, it's impossible for water to get behind the liner and that what we heard last time with looking for bubbles in the liner is probably not real good acceptance criteria. So we're looking forward to hearing how that's turned out.

One other had to do with the fatigue cycle estimates and the historical fidelity. We got a glimpse of the histograms, but we would ask for a little more explanation on that, a brief explanation of why we think ten years is a good -- the last ten years is a good predictor for future reactor vessel performance. And we had noted that our RDNDT would have exceeded the acceptance criteria and we're relying on the new ones to carry us there.

There were a few issues with Boral and some of the Unit 1 and 2 differences. I think the chairman noted that we received a letter and if staff is inclined to comment on that, we'd be interested in hearing what you have to say.

At this point, I think I'll turn it over to Brian Holian. Thank you.

MR. HOLIAN: Thank you, Dr. Bley and Chairman, good morning, ACRS members. My name is Brian Holian and I'm the Division Director for License Renewal. I'll just do introductions and a few introductory comments and then turn it over to the licensee for their presentation, followed by staff's presentation.

To my right is Dr. Sam Lee, Deputy Director, Division of License Renewal. To his right is the Project Manager for the Beaver Valley license renewal, Mr. Kent Howard. I'd also like to highlight just three members from Region 1 that are here today. Behind me is the Branch Chief of Division of Reactor Projects for Beaver Valley and that's Dr. Bellamy, Ron Bellamy. We also have the Senior Resident Inspector from Beaver Valley, Dave Werkheiser. And we also have a BRS Inspector who also will be heading on soon to

Indian Point as Resident Inspector, Ajo Ayegbusi.

We also have other branch chiefs and technical staff members that you'll hear from in response to questions during the staff presentation.

I would like to just highlight two items. There was one open item at the Subcommittee meeting in the draft SER and as you mentioned that was submerged cables and the issue of them being wetted or submerged historically. You'll hear from the licensee and us on that resolution of that issue.

Also, we had an issue, as you mentioned, that got quite a bit of discussion at the Subcommittee and that was the containment liner degradation first found in the 2006 steam generator replacement, exterior, some corrosion found in the exterior aspects. Following that Subcommittee meeting and that outage, you'll hear about it today. There was an issue identified during the outage of through-wall on the liner and you'll hear the root cause of that issue and what the licensee has done and also commitments they've made both in response to the exterior-type corrosion and this interior corrosion that did go through-wall.

On that issue, the staff did receive a letter from Citizen Power back in May responding to that issue and saying that the draft SER should be expanded to include aspects of this issue and we agreed with that letter. We responded in June and said we have sent additional RAIs to the licensee and we were further reviewing that issue.

As you mentioned just yesterday, July 7th, we received a letter this morning. We received another letter from Citizen Power stating that they understand what the licensee has committed to and their response for additional information, additional UT and just summarizing that letter quickly for the Committee, the two main items I got out of it is one, there's a commitment to do expanded UT. The letter takes issue with the timing of that. It's to be done before the period of extended operation and the letter basically says the sooner the better.

The second issue is the number of UT samples. Seventy-five, one foot by one foot areas were proposed and accepted by the staff. And there is some issue with the randomness of those, how you pick that sampling criteria. The licensee has proposed more of a smart sample and I think the letter takes issue with one, how you're doing that sampling and two, the amount that should be done based on the root cause. If you would exclude that issue, their issue would be that you would have to do more because you've had an instance where through-wall has come through.

So that's a quick summary of the letter. The staff will have to respond in writing to that letter and our technical experts haven't gotten all the way through it. We just received it this morning, but we'll be able to respond verbally to parts of that.

With that, I'll turn it over to Beaver Valley and Pete Sena, the Site VP.

MR. SENA: All right, thank you, Brian. And good morning.

Mr. Chairman, distinguished members of the Committee, thank you for the opportunity for Beaver Valley to present its application for license renewal. I'm Pete Sena, I'm the Site Vice President at Beaver Valley.

With me to my left is Cliff Custer. Cliff is the project manager for license renewal. Then there's Mark Manoleras. Mark is the director of site engineering at Beaver Valley. And we also have John Thomas. John is our senior technical lead for license renewal. Additionally, in the back we have members of the core license renewal team and members of the Beaver Valley staff that are available to answer any specific questions that may come up through the Committee.

Safe, reliable operation has been the priority at Beaver Valley for the last 33 years. Today Beaver Valley's safety record is one of the top in the industry and that's noted by our top decile metrics with respect to INPO index.

Our management of active components has been absolutely improved over the last 33 years of operation through PM programs, through critical spares, through corrective and elective maintenance, but as we're all aware license renewal hinges on our ability to manage passive components.

This morning, we'll have the opportunity to discuss, as Brian talked about, recent operating experience at Beaver Valley. From my viewpoint good news is not there are no problems. Good news rather is you're identifying your issues and you're correcting your problems, your issues, rather, before they become problems.

As we'll discuss with our containment liner activities, we believe that we are effective with our inspection program. We have corrected the deficiency and we've properly adjusted our going-forward actions.

With that, I'll turn it over to Mark Manoleras. Thank you.

MR. MANOLERAS: Thanks, Pete. What we'll today is we'll discuss a short description of the site. I will review the overall license renewal project. We'll discuss our open item resolution associated with inaccessible medium-voltage cables. Also, Cliff will discuss some of the subcommittee follow-up items you heard discussed before. This will include recent OE associated with MRP-146 and some of the inspections that were completed, and also some recent OE associated with our Unit 1 containment liner. We'll also provide an overall summary of the project.

We had already heard about his site description. Beaver Valley again is a two-unit, 3-loop Westinghouse PWR, 17 miles west of McCandless on the Ohio River. It's owned and operated by Ohio Edison and Toledo Edison, part of the First Energy Nuclear Generation Group.

Beaver Valley went commercial in 1976 and Beaver Valley Unit 2 in 1987.

I'll now turn it over to Cliff to discuss the license renewal project.

MR. CUSTER: Thank you. The license

renewal project, the Beaver Valley core team remained engaged with the industry. We attended several of other nuclear industry audits and inspections. Remained engaged with the NEI Working Groups, and of course the NRC meetings.

In addition, the application received independent assessments by an industry panel, our own site QA, an industry peer-review group, and the FENOC Corporate Nuclear Review Board.

Our methodology was consistent with NEI 95-10. From the very beginning it was our project intent to maximize Gall consistency. As you heard, we're nearly 92 percent of the AMR line items are consistent with GALL.

Our open item was identified in the draft SER and the subcommittee meeting on February 4th on inaccessible medium-voltage cables. I'm pleased to say that we've closed that open item. The method that we used to close the item was recognized that we needed to modify our Age Management Program for one that was more consistent with GALL.

We offered and provided the new commitment, the commitment of three parts to development a methodology to demonstrate the cables will continue to perform their intended function, minimize exposure to significant moisture or replacement of the cables. Our current priority is on minimizing the exposure to significant moisture and we're working in that direction.

With respect to some of the subcommittee follow-up items that you heard Chairman speak to, we had some recent operating experience during our spring outage in 2009 with respect to the MRP-146 inspections. MRP-146 is Materials Reliability Program.

VICE CHAIR ABDEL-KHALIK: Excuse me, on the previous slide, does the water in these manholes ever freeze?

MR. CUSTER: We have seen no instance of freezing in these manholes.

VICE CHAIR ABDEL-KHALIK: So the cables are never exposed to freeze-thaw cycles?

MR. CUSTER: I would ask Brian Murtagh to talk about that.

MR. MURTAGH: Good morning. I'm Brian Murtagh from Design Engineering. No, there's been no evidence of a free-thaw cycle.

VICE CHAIR ABDEL-KHALIK: Okay, thank you.

MEMBER BROWN: I have one other question on the cables also. In the subcommittee meeting, as a result of this discussion also, there were three things you were going to do. But in that subcommittee meeting you identified that you all were going to provide documentation to show those cables were designed for submerged operation and I didn't know whether anything else had been supplied along that line. I hadn't seen it. It's not reflected in these three action items.

MR. CUSTER: Brian, would you like to comment on that, please?

MR. MURTAGH: Yes, we can. During the subcommittee, we provided information that the cables were suitable for the environment and we did provide the staff the previous information regarding the cable constructions and the vendor letters that describe the cable. However, we have since come to an understanding that cables need to be more than suitable for the environment. They need to be qualified for the environment. Therefore, the localized environment for these cables has to be consistent with qualification and therefore we need to eliminate the submerged conditions.

MEMBER BROWN: Okay, that's a nuance on the word suitable like qualified?

MR. MURTAGH: Yes.

MEMBER BROWN: Okay, thank you.

MEMBER MAYNARD: Point of clarification. You're not doing all three of these. These are ors.

MR. CUSTER: These, in fact, are ors. Thank you for the clarification.

As a subcommittee follow-up item, as I said, in the spring of 2009 this year we had recent operating experience with respect to our implementation of MRP-146. MRP-146 is a Materials Reliability Program, guidelines for inspection of reactor coolant system branch lines for thermal fatigue.

We had made commitment, our commitment 31 for Beaver Valley Unit 1. It happened to be 32 for Unit 2. At Unit 1 in that outage there were 13 piping locations that were screened in as susceptible. All those locations were examined during our 1R19 spring outage. We identified on one line which happened to be the alpha loop drain line, a two-inch diameter line, nondestructive indications on that line.

The probable cause is in alignment with what was expected from MRP-146, thermal fatigue. However, we still have metallurgical confirmation pending to confirm that that is, in fact, the case. The pipe was replaced that contained the indication.

MEMBER POWERS: The screening analysis examined those things that are screened in, go back and redo the screening analysis now once you have an indication?

MR. CUSTER: I'd like Steve Buffington to talk about our methodology there.

MR. BUFFINGTON: I am Steve Buffington from Design Engineering. Sir, I'm not sure I understand your question.

MEMBER POWERS: The screening analysis, you find some things, then you go in and you find an NDE indication on one of those things you screened in. Doesn't that affect your acceptance criteria for your screening?

MR. BUFFINGTON: The screening for this is based on geometry and operating conditions. And we predicted -- well, we indicated that these 13 locations might be susceptible to thermal fatigue and those were the locations that were inspected. There are follow-up activities along MRP 146 and they include analysis that determines what the severity of thermal cycling would be at the screened-in locations. And then incorporation of that into design analysis, along with the other thermal transients that are occurring. And depending upon what your results of that detailed analysis are, identifies when you would do follow-up inspections and what further actions we would take.

MEMBER POWERS: So you don't -- I mean you set some threshold for your thermal cycling to do your screening analysis. That threshold is not intended to find anything?

MR. BUFFINGTON: No, that threshold is not depending on results of inspection.

MEMBER POWERS: It seems like it ought to be, doesn't it? I set a threshold based on something. I find indeed things are exceeding that threshold. Shouldn't I set a more restricted threshold?

MR. BUFFINGTON: Well, these locations did exceed the threshold which is why for the screening which is why we went and looked at them.

MEMBER POWERS: And if you subsequently find an NDE indication, isn't the threshold maybe a little too generous?

MR. BUFFINGTON: If I may, it's a question with respect to the threshold of the acceptance criteria for the NDE, for the UT exams, or the screening for scoping in?

MR. SENA: If I may, Steve, wouldn't the fact that we found something consistent with our screening confirm that the screening was appropriate?

MR. BUFFINGTON: Yes, I believe that's correct. I think if we had found indications in something that screened out --

MEMBER POWERS: But you never looked if you screened it out.

MEMBER ARMIJO: You inspected 13 locations that were screened in. Of those 13, you found indications only on one location -- the screened criteria were not -- were I would say somewhat conservative, yes, that you -- if you found 13 out of 13 with defects, I would have said you better reset your criteria, because the threshold is lower than what --

MEMBER POWERS: What is your probability of making that type two error? And somewhere around 10 percent probability which is what you would have here is a little high, I think. I don't know what the probability is on your screening, but I would assume your screening has the likelihood of me having a flaw and I screened out things like one percent or something like that. It would be my screening type. I don't know what theirs is.

MR. MANOLERAS: Yes, I definitely -- this is Mark Manoleras, the Engineering Director of Beaver

Valley. I definitely understand the question. MRP-146 provides pretty solid guidance on how to select the locations. The analysis is then performed. The identified locations are then screened in. The inspections are then performed, and then the follow-on actions are identified.

I believe that we are definitely following in accordance with the guidance of MRP-146. I definitely understand your question. We've entered that into our corrective action system in doing some additional evaluation additionally.

MEMBER BLEY: Let me ask a related question. It's been kind of hinted at and that would be do we know what that -- the things that are screened out, the screening procedure, does it have in mind a likelihood of gauze being in the places that are screened out? Is that the screening criteria?

MR. CUSTER: Steve, would you like to explain that?

MR. BUFFINGTON: The screening criteria, this was put together as part of the MRP-146 program and that's basically screening us on geometry and the flow in the loops and how you would develop a thermal cycling within that unisolable branch line. That process is all based on testing in the industry and also industry experience including other cracks that have occurred on the unisolable branch lines.

MEMBER SHACK: It seems to me that the screening criteria is called in question if you have a failure or some indication that is found in a component that would have been screened out. That would be the criteria. The screening criteria is appropriate if after you do the examination you find an indication in something that was screened in, but the reverse is not necessarily logical in my mind.

MEMBER ARMIJO: Just for a little more detail, of these 13 locations that were screened in, was there any kind of ranking of the most likely and did that correlate with the one location where you found the defect or the indication? In other words, were they all viewed as an equivalent risk or was there some --

MR. BUFFINGTON: Yes, I'd like to answer that. There are basically two configurations that we were including and that's when you branch off of the top of the loop that's considered an up horizontal configuration or a down horizontal configuration.

The location we had the indications was down horizontal, and in this particular instance there was nothing unique about this where we would think that we had flaws on this loop versus the other two loops.

We did inspect 100 percent of our down horizontal locations and did not find indications in the other five locations.

MR. CUSTER: If I could, I would like to bring Dennis Weakland to the microphone.

Dennis, would you care to talk to us a little bit about the inspection criteria developed by the industry for MRP-146?

MR. WEAKLAND: I am Dennis Weakland. FENOC Materials Corporate. I'm also chairman of the MRP IIG integrations group that produced this document for the industry under EPRI.

The 146 examinations were developed analytically over the past several years and experiences we've seen with small-bore and nonisolable components off the RCS loop because the industry saw an issue. The industry took a voluntary action. This is all of the MRP-146 documents were done under the NEI initiative 03-08 to which our outside of code, nonmandated. These are initiatives that the executives imposed upon themselves to take on.

These inspections that were performed at Beaver Valley were the first round of inspection programs for the industry. We're just now getting through the completion of that by the end of this calendar year.

And what you're seeing is part of the feedback mechanism that we have, the metallurgical work being done will be fed back into the criteria to assess the analytical work, was it correct, and we generally will revise our guidance as we have done with MRP-139 for Alloy 600. We had to revise that because we found things in the field. That's the purpose of the guidance. It's go out, get ahead of the issue, find the issue before we find failures in the industry and it was -- it did exactly what it was supposed to do. We found thermal fatigue, what we believe to be thermal fatigue, prior to it becoming a failure. It was being proactive in the materials perspective. That's the purpose.

MEMBER BLEY: Thank you. Let's go on.

MR. CUSTER: Thank you, gentlemen. Okay, in the next subcommittee follow-up item, on the screen right now are early containment construction photos that were previously requested from the subcommittee. The picture demonstrates in situ liner construction and the degree of rebar density involved in the design.

Next slide please.

The liner design itself, carbon steel liner. Nominal thickness on the floor is one quarter of an inch. Three-eighth's of an inch, nominal thickness on the wall and a half inch on the dome.

Insert plates are installed into the liner. Those are 5/8ths to inch and a half thick. They have separate studs so that any large loads are transferred to the concrete of the liner. There are overlay plates attached to the liner for very light loads such as cable trays and so on and penetration strengths for the loads directly to the concrete in the wall.

Now the studs on the liner on 12-inch centers and the liner itself is a leak-tight membrane. It performs no structural function.

COURT REPORTER: Sir, that's your paper on the microphone.

MR. CUSTER: Thank you very much. Continuing on, as we discussed previously in the subcommittee meeting in 2006 --

VICE CHAIR ABDEL-KHALIK: Just a clarification, if I may, this is a leak-tight membrane. What is the functional purpose of the liner?

MR. CUSTER: John, would you like to discuss this?

MR. THOMAS: It's the fission product barrier. It's a gas membrane for containment to retain fission products after an accident.

VICE CHAIR ABDEL-KHALIK: So leak-tightness is an important performance measure for that functional requirement of the liner?

MR. THOMAS: Containing the fission products following an accident, yes.

VICE CHAIR ABDEL-KHALIK: Thank you.

MR. CUSTER: Returning back to -- as we previously discussed in the subcommittee meeting in February, in 2006, during our 1R17 outage for steam generator replacement, during hydro-demolition, removal of the concrete for the 20 by 20 opening for the steam generator, we exposed the backside of the land. We identified three areas of corrosion on the concrete side of the exposed liner. None of these areas were, in fact, through-wall. The areas were randomly spaced within that 20 by 20 area. There was no necessarily any pattern.

In 2009, this spring, during the scheduled visual inspection in accordance with the IWE code, we identified paint blisters with some rusting.

Subsequent cleaning revealed the primer coat, in fact, was blistered, in a small through-wall flaw, approximately one inch by 3/8ths of an inch.

The volumetric UT exam determined the extent of corrosion around the flaw was an area around two by five inches, two inches by five inches.

MEMBER POWERS: You call this a small flaw, but if I compare that flaw size to your design basis leak rate, I think it's not small.

MR. MANOLERAS: Bill Etzel, can you please talk about that?

MR. ETZEL: Yes, this is Bill Etzel, the lead PRA engineer at Beaver Valley. We looked at the risk significance of the hole and looked at the equivalent diameter which would be about a .7 inch circular hole and compared that to our definition for large early release frequency which has a minimum diameter of two inches. So we were a lot smaller than our required minimum granule size. So any release would be small early release.

MEMBER POWERS: I have no idea what you're talking about. If I compare this to your design basis leak rate that's a hole of what, roughly two millimeters in diameter would give you your design basis leak rate. And this is enormous compared to that.

MR. ETZEL: We also looked at industry OE. North Anna had a similar containment liner hole back in 1999 and they did a localized pressure test. They had approximately a quarter inch hole diameter. So we took their test results and scaled them up by the ratio of the areas and then took our as-found type A test leakage and added those two leakages together to come up with a total estimated leakage through hole. That value was less than or maximum allowable containment leakage rate.

MEMBER POWERS: I am surprised. Let's put it this way. I don't know what your design basis leak rate is, but I'm guessing it's around .1 percent per day. And the question is do you now come into violation of 10 CFR Part 100 doses at the site boundary for the design basis source term going into this plant? And that seems to be offered.

MR. MANOLERAS: Bill, go ahead and address that.

MR. ETZEL: Yes, our design basis is .1 percent containment error mass per day. You have to factor in that after we took away the containment liner, the concrete behind the liner was in good condition. So it didn't have a through-wall through the concrete. So that would also provide additional barrier to leakage.

MEMBER POWERS: So you're taking some credit for fission product continuation by the concrete?

MR. MANOLERAS: No, what Bill is trying to explain is that the results were conservative to the Part 100 limits. In addition, we did not take credit to the concrete behind the liner.

MR. SENA: We took no credit for the concrete. We took no credit. That's just additional conservatism them.

MR. MANOLERAS: We took credit for North Anna's test which had the concrete --

(Laughter.)

MEMBER POWERS: One would hope --

MEMBER BLEY: Tell us about this two by five inch flaw. You said it's equivalent to a .7 inch diameter circular hole. So it really wasn't two by five?

MR. CUSTER: Let me comment to that, please. The opening was one inch --

MEMBER BLEY: That was the one by one by 3/8ths.

MR. CUSTER: One inch by 3/8ths.

Horizonal there was some loss of wall.

MEMBER BLEY: That was the extent of corrosion. Okay, thanks.

MEMBER ARMIJO: Was this through-wall hole detected by any leak rate test, routine testing or periodic testing or was it only detected by the blister and subsequent exam?

MR. CUSTER: It was detected by the blister. And the subsequent exam that followed it up.

MEMBER STETKAR: Do you have a risk-informed ILRT frequency now in place at Beaver Valley?

MR. ETZEL: We had a risk-informed one time extension, but it's no longer risk-informed.

MEMBER STETKAR: When is the last time you did an ILRT?

MR. SENA: That would have been after the steam generator replacement outage.

MR. CUSTER: 2006.

MEMBER STETKAR: Thank you.

MEMBER CORRADINI: And that was -- just to be clear, that was before you found this?

MR. CUSTER: Correct.

MEMBER POWERS: That would suggest that this corrosion progresses very, very fast.

MEMBER CORRADINI: Or, just another way of saying it, or if I understood how he explained your interpolation, it was part of it and it was below the limit.

MEMBER POWERS: Mike, a hole like this will never make the integrated leak-rate test.

MEMBER CORRADINI: Unless I misunderstood his explanation, they took the --

MEMBER POWERS: I didn't understand his explanation at all, so --

(Laughter.)

MEMBER CORRADINI: Okay, but as he explained it, he took the North Anna results, scaled it with area at their IRLT which is 100 and something percent of design pressure and then showed that was well within their leakage.

So for the leak rate part of it they are taking credit of the containment concrete.

MEMBER MAYNARD: It does not surprise me that you pass on IRLT with a hole of that size in the liner as long as the concrete is good behind it which is what I think the condition was.

MEMBER BLEY: Thank you.

VICE CHAIR ABDEL-KHALIK: How comfortable are you that this is the only sort of location where you have wastage in the containment liner?

MR. MANOLERAS: We believe that the programs we put in place will identify these locations prior to us exceeding any of the design limits.

VICE CHAIR ABDEL-KHALIK: But as of today, do you know the state of the liner, other than the fact that you have identified this particular hole?

MR. MANOLERAS: We have just completed our IWE inspection of this outage and we have successfully completed that IWE inspection. Additionally, the flaw that we identified we repaired and performed a leak test on, so yes, yes, we believe with that IWE inspection and the repair of that location in the liner that our liner meets the requirements. That's correct.

MR. SENA: So if I may, the IWE code inspection, three inspections over a ten-year interval requires 100 percent visual inspection of the accessible containment liner within the containment structure. We completed that 100 percent inspection this outage. This was the one blistered location we did identify.

We had the Type A test as we stated back in 2006. This was the code inspection which identified the blister which we then cleaned and removed the rust away to identify the through-wall condition. So I believe what's important is now we found a problem, now you have to adjust your going-forward inspection plans.

MEMBER SHACK: Let me ask a half-related question. In the early days at Beaver Valley that was a sub-atmospheric containment and the pressure during operation was about ten pounds absolute.

MR. SENA: Correct.

MEMBER SHACK: You chose to utilize the alternate source term which allowed you to reduce the amount of backing in the containment. What pressure do you -- you're still negative?

MR. SENA: Still negative.

MEMBER SHACK: What pressure do you operate at now?

MR. MANOLERAS: About a half a pound sub-atmospheric.

MEMBER SHACK: Good. At ten pounds absolute if you had a significant hole in the containment, you could tell by the pump out rate.

MR. SENA: Absolutely.

MEMBER SHACK: At a half a pound, I doubt that you could tell, right?

MEMBER RAY: Wait a minute, all of this reference to the test results inevitably winds up with the concrete masking what the leak rate is going to be from the membrane. The membrane is supposed to prevent leakage from a design-basis accident.

MEMBER SHACK: Right.

MEMBER RAY: At which point the condition of the concrete can't be taken credit for. So I guess I just think that the idea that the leakage is going to be small from a small hole, from a hole this size, as small as Dan says, in the design-basis conditions isn't logically supportable because the concrete, you can't -- you, yourself said, you can't take credit for the concrete and the reason is because it's condition in the design-basis event can't be predicted, can't be credited. The only thing you can credit is the membrane itself.

MEMBER SHACK: From a deterministic basis, you're correct. From a probabilistic basis, which is what they use and can take credit based on --

MEMBER RAY: I don't think so.

MEMBER SHACK: Well, that's the way it is.

MEMBER RAY: That's not right.

MEMBER SHACK: I'd like to get an answer to my question that I asked before.

MR. MANOLERAS: Bill, why don't you take a shot at that question?

MEMBER SHACK: Could you tell? The answer is probably not, right?

MR. MANOLERAS: Probably not, Jack.

MEMBER SHACK: Okay.

MR. SENA: Jack, as a former senior reactor operator at a half pound, I'd agree, probably not.

MEMBER SHACK: Probably not.

MR. SENA: If they trend it long term and if you're particularly looking for that, perhaps.

MEMBER SHACK: I even doubt that because of the temperature difference. Okay.

MEMBER BLEY: Before we leave this, I've forgotten, what kind of corrosion was this and where did it occur?

MR. CUSTER: This corrosion was a localized corrosion.

MEMBER BLEY: Was it inside or --

MR. CUSTER: From the outside of the concrete side of the liner to the inside.

MEMBER BLEY: From the concrete, so between -- and what kind of corrosion was it?

MR. CUSTER: It was a pitting attack.

MEMBER BLEY: So there was moisture in there?

MR. CUSTER: Yes.

MEMBER RAY: It was a piece of wood that was the site of corrosion.

MR. CUSTER: If I could, gentlemen, my next slide will answer some of those questions and we can do a follow up with that.

MEMBER CORRADINI: Before we go on to what caused it, I guess I heard between Jack and Dana and Harold three different opinions about whether you can or cannot take credit of the concrete for the design basis. So I'm still not clear if you can or cannot.

MEMBER SHACK: Cannot.

MEMBER CORRADINI: So then Dana's question is operative, that you can't use this sort of analysis to estimate your leak rate.

Is that correct?

MEMBER SHACK: From a risk standpoint, yes? From a design basis standpoint, no.

MEMBER RAY: The reason I disagree with Jack on the risk standpoint is the risk model for the behavior of the concrete in the design basis event I think has got to be explored before you claim, take credit for the concrete on a risk basis.

MEMBER POWERS: We don't know how to do it is the problem.

MEMBER RAY: That's my point.

MEMBER BLEY: This is a key issue for us, but I think we need to go ahead, because we're almost out of time and then we want to hear the rest of what you have to say. So please go ahead.

MR. CUSTER: Okay, continuing on with the description of the screen occurrence here, we found wood immediately behind the liner. That analysis confirmed that there was moisture in the wood. The corrosion was attributed to this wood in contact with the liner in the presence of moisture. As we said, our concrete was found to be in good condition and we replaced the sectional liner.

Our corrective actions with respect to this event, of course, a follow-up UT of the replaced area during the next Unit 1 outage. We did do baseline of the replaced area. We have planned additional 100 percent IWE visual inspections for the next Unit 1 and 2 refueling outages.

We will maintain our schedule for the normally-scheduled exams for the final outage, and we intend to do supplemental volumetric inspections on both liners prior to entering the period of extended operation. That is a random inspection on these areas in accordance with the guidelines from IWE that provide a methodology similar to give us a high-level confidence on those areas.

MEMBER BLEY: Can you explain for us a little bit how you do this volumetric examination?

MR. CUSTER: The UT examination?

MEMBER BLEY: No, I'm okay.

VICE CHAIR ABDEL-KHALIK: What is your tech spec limit on the containment leak rate?

MR. CUSTER: Bill, do you have that information?

MR. ETZEL: This is Bill Etzel again. As I stated previously, our containment tech spec leakage rate is .1 percent of the total air weight per day. And that equates to about 6,831 standard cubic feet per day.

MEMBER SHACK: That is at design pressure.

MR. ETZEL: That is at design pressure.

MEMBER SHACK: Through SDP.

MEMBER ARMIJO: But that is an integrated leak including the liner and the concrete and I don't believe you have a capability of just what's leaking between from past the liner. So I don't know how else you could measure?

MR. MANOLERAS: That's correct.

MEMBER RAY: You identified a mechanism

for the through-wall corrosion wood debris. Was there ever any mechanism identified for the prior observed corrosion when the steam generator replacement was made?

MR. CUSTER: In 1R17 when we did the steam generator replacement, the section of concrete was removed by hydrodemolition, high water pressure. As a result, we searched the concrete debris field, but found nothing as a result.

MEMBER RAY: Thank you.

MEMBER ARMIJO: Just order of magnitude, what was the extent of and mechanism of corrosion in that large area? Was it pitting or just generalized thinning or what?

MR. CUSTER: There was some generalized and some pitting attack as well.

MEMBER ARMIJO: Okay.

VICE CHAIR ABDEL-KHALIK: I asked earlier about the functional purpose of the liner and you stated that it's a leak-tight membrane. How is that functional requirement -- accomplishment of that functional requirement is attained? Can you measure the leak rate of the liner in and of itself?

MR. MANOLERAS: I can answer that question. The liner performance is verified by several facets. Every ten years we perform a type alpha test, type A test. We pressurize the containment and measure that leak rate. Then three times over that interval, we basically do an IWE inspection over the 100 percent of the visually accessible areas of containment. Those are the two manners in which the code requires you verify the liner performance.

VICE CHAIR ABDEL-KHALIK: So you actually measure the leak rate of the liner in and of itself?

MR. MANOLERAS: You cannot.

MR. SENA: It is the entire containment structure.

MEMBER RAY: The IRLT mostly measures the leakage of penetration.

MR. SENA: That's correct.

MEMBER RAY: And so that's what you're measuring and the visual inspection is used to assure the continued integrity of the liner. That's the way it works.

MR. SENA: That's correct. Well, again for 10 CFR per the code then you also have your type bravo testing of your major access areas or containment airlock for example. And then of course, you have your type C testing of your individual penetrations.

MR. CUSTER: Okay, so continuing forward, I'd like to turn it back over to Mark if there are no further questions.

MEMBER ARMIJO: I guess, you know, having looked at your documentation, your whole inspection approach going forward is based on the assumption that the mechanism is caused by this wood, moist wood in contact with the liner causing localized failure over time.

And you're going to have some random UT inspection and 100 percent visual to give you some indication of whether there might be the same mechanism operating elsewhere. And really, the only thing that you -- you don't know where the wood, where other pieces of wood might be, so you're going to rely entirely on either by chance that your UT will find a location or the visual will be reliable, that you'll always form a blister that tells you that liner is pretty much --

MR. CUSTER: At that point in time something has gone through. Keep in mind that the methodology for choosing the random location is in alignment with the statistical methodology providing 95 confidence level similar to what's used in the IWE code.

MEMBER CORRADINI: Can you repeat that last part? I'm sorry.

MR. CUSTER: The methodology that we will be using to choose these random locations is consistent with the methodology to provide 95 percent confidence level to identify these areas similar to that used in the IWE code.

MEMBER CORRADINI: Okay, so what if I change 95 to 99, what would the number of samplings be from?

(Laughter.)

MR. THOMAS: We would need to calculate that, but it would be very substantial.

MEMBER SHACK: From 90 to 95 it goes from 25 to 75 and so you can sort of take the slope.

MEMBER CORRADINI: Let me just ask one more question. So when you do the visuals, you essentially photograph -- I'm still trying to understand how you do the visuals. You photograph certain blocks of containment?

MR. CUSTER: What I'd like to do is ask Dave Grabski our IS individual to describe how he does those inspections and respond to the question.

MR. GRABSKI: Yes. This is Dave Grabski.

I'm the ISI program owner at Beaver Valley. How we perform those examinations are we use a systematic approach based on the I-beams in our containment. So we'll ask the inspector to measure or examine visually between these two. If he finds anything, of course, that's the way he references it. So it's a visual. He wouldn't necessarily take pictures or a video of it, unless of course, there was an indication and then we would take pictures.

MEMBER ARMIJO: Just let me ask, is he going to report a one-inch diameter blister or just a three-inch diameter blister or any blister?

MR. GRABSKI: Any anomaly whatsoever, whether it's a blister or whether it's a scratch, scrape.

MEMBER CORRADINI: And then you go in and do the additional inspection?

MR. GRABSKI: right.

MEMBER CORRADINI: On top of your sampling inspection.

MR. GRABSKI: Well, if we found any kind of anomaly, we would ask a qualified Code VT examiner to come and take a look at it before we did anything.

MR. SENA: If I may, I think it's important just to kind of summarize and put this all together. So we've done the Type A test per the code every ten years. The code then requires the visual inspection three times per interval. We believe that visual inspection was effective and that we did identify a deficiency. We correct that particular deficiency. That is corrected.

Now what's important then is so what do you do going forward? And we believe that going above and beyond the current code requirements by performing additional visual inspections, by essentially short cycling during the supplemental inspection next outage for both units and then doing the additional volumetric exams with the 95 percent confidence criteria is appropriate for the actions going forward.

MEMBER SHACK: Describe for me, this is before the period of extended operation which is how many outages?

MR. SENA: Well, the next visual exam will be done next outage for both units.

MEMBER SHACK: When do I start the UTs?

MR. SENA: The UTs, Mark?

MR. MANOLERAS: Yes, and again, the commitment was made for license renewal. That's why it was submitted prior to the period of extended operation. We expect to complete in a very timely manner. I will work with our project manager and our project owner to get those done in a very soon subsequent outage.

MEMBER CORRADINI: So the answer to his question is what though?

(Laughter.)

MR. SENA: We're scoping our current outage right now.

MR. MANOLERAS: The availability of resources, making sure that we have the criteria set. Make sure that the random locations are set. So we're in the process of working through that.

MEMBER ARMIJO: But when you do the next outage, the next inspection, and let's, for example, you find some blister, inch, two inch, whatever, will you do UT then? Will you do something else? Or just say hey, we found a blister and we'll do UT a few cycles from now?

MR. CUSTER: Our methodology is pretty much consistent. As a matter of fact, it's proceduralized. If we find any blister, as David said, the first thing that we do is a VT-2 inspection, determine the extent of what's there. We follow it up with a UT, if we expected that there was any primer coat delamination or anything of that nature, rather than a top coat delamination.

MEMBER CORRADINI: I'm sure this was said in the subcommittee, but just remind me, what is the level of detection if you do a UT in terms of percentage of through-wall? When do you start seeing something that worries you? What's the indication? Is it 10 percent of through-wall? Is it some fractional amount, half of it in terms of pitting? If you were to have done the UT on what you found --

MR. GRABSKI: Yes, this is Dave Grabski again, we would obtain from design engineering a screening criteria for that thickness. If it was above that thickness and acceptable, we certainly would trend it and track it. If it was below, then we'd have to take the necessary corrective actions. But we would go in there with a number from our design engineering based on where the indication is located.

MR. CUSTER: I think it's important to point out here that we're talking of pitting/corrosion type of attack where the criteria would be developed based on the diameter of the pit, the depth of the pit.

MEMBER CORRADINI: Right.

MR. CUSTER: So it's not like it's a uniform corrosion where there would be a number.

MEMBER CORRADINI: Right.

MR. CUSTER: That would be --

MEMBER CORRADINI: What I guess I'm asking and I'm sure you answered it, but I just didn't understand the answer, at what level does the signal start worrying you? I think I heard you say well, it kind of depends, but I'm trying to get a feeling for what does that imply in terms of a physical pit size that you start going across a boundary and then start performing some sort of action other than watching it.

MR. GRABSKI: The liner plate is fairly consistent. It will have some low points here and there based on what we've seen, but anything less than ten percent would start getting our interest.

MEMBER CORRADINI: Okay, thank you.

MEMBER SHACK: What is the actual code requirement on this liner since it has no structural function. This is not like a steel containment where I would do an analysis, strength analysis. What do I do and what is the requirement here? Do I just have a remaining ligament?

MR. GRABSKI: We did an evaluation three years ago and again I'm talking off the top of my head here. I think the general wall thickness requirement, that's general, was in the 140 range. Anything else, if you had an indication one inch and small, it could down below 100, maybe to the 40 mil range. Don't hold me to that. Again, I'm just recalling what we had based on the evaluation we had in 2006. It is a membrane if its membrane thickness is localized.

MEMBER SHACK: What part of the code do I use to do that analysis? Is it the same analysis that I use for the steel containment?

MR. MANOLERAS: Steve.

MR. GRABSKI: I will speak from Section 11. It's going to give it to the owner to do it, as far as construction code.

MEMBER SHACK: The ones you did have you lost about half the wall in the worst case and you had a pit down to .33 depth, which is getting pretty deep.

MR. WEAKLAND: This is Dennis Weakland. Generally, the way you would do these types of examinations is very similar to the way you would do any pitting evaluation for buried piping, other piping lines and the rest because pits tend to be very localized. The smaller the pit, the thinner the wall that you can handle because it has supporting structure around it.

MEMBER SHACK: In that case, the pipe wall has a structural function. I know how to do that analysis.

MR. WEAKLAND: We would look at it --

MEMBER SHACK: This has some structural function, does it?

MR. WEAKLAND: We would look at the design pressure at accident, 45 pounds, and say what do I need, wall thickness, to withstand that based on the size of that opening. So if the pit --

MEMBER SHACK: So I'm doing an analysis with a concrete backup?

MR. WEAKLAND: No.

MEMBER CORRADINI: That's what I think they were saying.

MR. WEAKLAND: You assume the concrete doesn't exist. You're essentially allowing it to expand.

MEMBER SHACK: And the .375 shell is going to take the design pressure?

MR. WEAKLAND: No. You don't assume it for the pressure across that membrane. That's what you're doing. You use the 3/8ths plate. You apply 45 pounds across a specific area.

MEMBER SHACK: A liner and a concrete shell.

MR. WEAKLAND: If you assume a 3/8ths

thick --

MEMBER SHACK: The liner needs nothing.

MR. WEAKLAND: And you have a pinhole, if you have 5 mils at the surface and you have something that is a tenth of an inch in diameter, doesn't need to be very thick to handle 45 pounds. If it's a half an inch in diameter, it's got to be thicker. If it's an inch in diameter, it's got to be thicker yet. that's how the analysis is. It's very similar to pitting corrosion on piping.

MEMBER SHACK: I think you're relying on the integrated leak rate test to determine the integrity of the liner. Visual examination does not show anything until your through-wall and if it is caused by moisture on the outside, and so in between picking it up as a visual and the periodic integrated leak rate test you're in sort of an area where you don't exactly know what the liner condition is.

On the other hand, there's a pretty good assurance that if you pass these tests, if you do the visual exam and detect a small hole, then the presumption is the hole will be small. But that's an assumption.

MEMBER BLEY: Gentlemen, are we done?

MR. SENA: If I may try to answer your

question again with respect to when are we going to do the UTs? It's important to note that the need to do these additional volumetric exams have been entered into our corrective action program. 10 CFR 50, Appendix Bravo Part TR16, of course, dictates timeliness, right? Prudence on our part also dictates the need not to wait until 2027, not to wait until 2016. So that's what we're evaluating right now as far as looking at resources, outage scopes scheduling, as far as when we can place it prior to the period of extended operation so the commitment was simply prior to the period of extended operation, but not to wait until 2016 or 2027. That's what we're looking at right now.

MEMBER BLEY: Okay, thank you. Thanks very much. Do you have more to close with?

MR. MANOLERAS: Just again we appreciate the opportunity to present the license renewal application to the ACRS today. Thank you.

MEMBER BLEY: Thanks very much. Thank you for your presentation. We'll have a little discussion come the end of the next presentation. Thank you.

We were a little bit longer than we were scheduled because that's of high interest to us. If we can move through the more routine things quickly and focus on the key points, we'd appreciate it.

MR. HOLIAN: Brian Holian again, Division of License Renewal, once again the staff's presentation will be made by Kent Howard, the project manager, assisted by Ron Bellamy, a branch chief from Region 1.

MR. HOWARD: Good morning. My name is Kent Howard and I am the project manager for the Beaver Valley Power Station license renewal application.

Today, we will present the results of the staff review of the application as documented in the Safety Evaluation Report.

To my right is Dr. Ronald Bellamy. Dr. Bellamy is a branch chief in Region 1. Dr. Bellamy will present a slide detailing the results of the June 2009 regional inspection that reviewed inaccessible medium-voltage cables and the containment liner issues.

Also with us in the audience are members of the NRC staff and of course, they're here to answer any questions that may arise.

Next slide.

This slide is an overview of some of the site information containment in the LRA. The applicant covered this pretty thoroughly in their presentation, so we'll just continue right on past this.

Recap of the February 2009 ACRS subcommittee meeting, the SER with open items was issued on January 9, 2009. There was one open item in the SER open item. It was the inaccessible medium-voltage cables. There were no confirmatory items. There were 249 RAIs issue. At the time there were 31 commitments for Unit 1 and 32 for Unit 2.

The addition of the number of commitments is that Unit 2 uses a wood pole electrical structures inspection program that Unit 2 does not have.

Next slide.

This slide is a summary of the follow-up items from the February ACRS subcommittee meeting. Those follow-up items are the inaccessible medium-voltage cables, the containment liner issue, the Boral, which was a new program and the metal fatigue/cycle count histograms.

For our presentation this morning, staff wanted to focus on those four items.

Subsequent to the subcommittee meeting there were six additional RAIS issued. We resolved open item 3.03.1.11-1 related to the inaccessible medium voltage cables. There was an additional committed added for both Unit 1 and Unit 2 which brought the number of commitments to 32 for Unit 1 and 33 for Unit 2.

The recent containment liner issue was addressed in the final SER which was issued on June 8, 2009.

Next slide.

Now we get into our inaccessible medium-voltage cable issue. During the staff's review there was one open item. The open item dealt with the inaccessible medium-voltage cables. During the aging management programs audit in March 2008, headquarters staff was concerned that inaccessible medium-voltage cables that had been submerged for a period of time may be degraded and may not perform the intended fund during the period of extended operation. The staff requested that the region follow up this item during their audit that was held in June 2008.

In this slide, I would like to point out that the SER with open items, inaccessible medium-voltage cable AMP was a plant-specific program. That program was revised to be consistent with GALL XI.E3. The applicant committed to either one of three options. They would either adopt an acceptable methodology that demonstrates cable will continue to perform their intended function, or they will implement measures to minimize cable exposure to significant moisture through dewatering manholes, they're going to pump them down, or they're going to replace the in-scope, continuously submerged medium-voltage cables with cables designed for submerged service.

MEMBER SHACK: Kent, you mentioned they're going to pump them down. Does that mean periodically inspect and pump them down when they find water, or are they going to install sump pumps? I didn't want to interrupt.

MR. HOWARD: No problem. In speaking with the applicant, they are going to install sump pumps with a level switch and right now they're trying to establish --

MEMBER SHACK: I just wanted to make sure they were going to have some sort of continuous process. Thank you.

MR. HOWARD: Are there any other questions on the inaccessible --

MEMBER BROWN: You used the word "designed." We had the nuance between suitable and quality. Does design mean qualified in this case?

MR. HOWARD: No.

MEMBER BROWN: Okay, so we're back to suitable again.

MEMBER ARMIJO: Well, that is a problem, isn't it? If it isn't qualified for that service by test, why isn't it acceptable?

MR. HOLIAN: This is Brian Holian. The answer is on number one that they would have to demonstrate, as we talked to the subcommittee, a cable that is designed, not suitable, is qualified for underwater, if they were to choose that method. If you remember the subcommittee, those members that were here, that was their original thought. They thought they had enough test data to do that. The staff said no, we don't believe you. And that discussion continued since the subcommittee and you heard the applicant say okay, we understand your position and they've left it as an option, should they go ahead and replace that cable and put it in as number one or convince us that they have done testing.

MEMBER BROWN: The answer is really design means qualified?

MR. HOLIAN: Yes, design means qualified.

MEMBER BROWN: All right, that resolves my problem.

CHAIR BONACA: One question I had from previous license renewals and other licensees, the issue for that was that the concern was cycling between the dry condition and the wet condition. That was the most challenging to the cabling. So could you address how the -- the alternative three would be successful.

MR. HOWARD: I would like to defer that question to Mr. Duc Nguyen.

MR. NGUYEN: My name is Duc Nguyen and I would like to address your question. The issue with the inaccessible medium-voltage cable, but water treatment phenomena. Probably the water would permeate the insulation during the cable energize, so you are right that most of the problem is dry and wet condition, due to the cable energized most of the time. So that's the problem with the issue inaccessible medium-voltage cable.

CHAIR BONACA: So I guess the cable is designed for submerged service would also be resistant to continuous alternation of drought and wetness?

MR. NGUYEN: Yes, but qualified to be submerged, we call it the submarine cable, they have the last sheet outside the cable would prevent the moisture to permeate the insulation and most of the cable that is installed in nuclear power plants, they are not qualified, but they qualified for the EQ local test, but they're not qualified for continuous submerged. That's why we did not agree with the applicant based on the test data provided to us that the EQ local test data which is not the submerged test data.

MEMBER SHACK: As I recall, these people have not just the wetting problem, they have some cables that are genuinely submerged all the time.

MR. NGUYEN: All the time, yes. And they are not separate cables.

MR. HOLIAN: This is Brian Holian, just to interrupt Duc, I think we have an electrical engineer representative and also Dr. Bellamy. I think the question also might be going to okay, you've had some periods now where they've been submerged. What has that done to the cable itself for continued operation or premature aging. The licensee has entered again -- Ron, you might want to mention the recent inspection where the Region went out with the Electrical Engineering Branch from Headquarters to look at the issue and kind of force the point on you have had a history of this, so the Region is looking at following up on their corrective actions for that.

Ron?

MR. BELLAMY: I can do that now, if you'd like. One of the Regions' Specialist Inspections was just done in June of this year. We had an electrical expert as well as a Region 1 manager accompanied by NRR technical support. This was done in light of an inspection sample with respect to problem identification and resolution and the team, these three individuals did look at the condition of the vaults. They did observe that although there was some moisture in one vault and measurable water in one of the other vaults, the vaults are periodically pumped down when water is observed in the vaults.

The licensee has committed, FENOC has committed to a long-term program of considering exactly how to ensure that the vaults stay drier, not dry, but drier, so there is not standing water in the vaults for extended periods of times. One of the options that they're looking at is to put a water sensor and then an automatic sump pump type system in probably two of the vaults. That schedule has not been set yet. Dave Werkheiser, the Senior Resident, and I will ensure that we continue our inspections in that area and we will document any results that come from those inspections in future inspection reports.

Now with respect to the medium-voltage submerged cables issue, we have not completed our evaluation of exactly how a licensee has done with respect to that yet. We have not done an exit interview with them. That exit will be held on July 22nd. Mr. Werkeiser and I will conduct that exit and we are looking at one potential finding with respect to design control. So we will monitor the licensees' corrective actions as they go forward from this point.

VICE CHAIR ABDEL-KHALIK: Now the licensee stated that they had never observed the water in those manholes freezing. Is there any physical reason to expect that in that part of the country that water that' stagnant in manholes would never freeze?

MR. BELLAMY: This is northern Pittsburgh area. I'm not aware of any. I've been at the Beaver Valley plant for five years now and I'm not aware of any instances where there's been water reported and freezing in these walls or any other type of contained water activity on this site. Obviously, the river there does freeze in chunks at times. But we have never seen any in our inspection activities of any water freezing there.

MEMBER BROWN: How deep are the manholes?

MEMBER SHACK: It depends on the depth

underground.

MEMBER BROWN: I was going to say the same thing, depending on the depth of the manholes, you go pretty far down. You can get a pretty stable temperature profile, 45 to 50 degrees --

MEMBER SHACK: And the depth in that area is roughly a foot and a half to two feet, at least these cables are like ten feet down.

MEMBER STETKAR: Ron, just to remind us, this topic actually is current licensing issue.

MR. BELLAMY: Yes, it is. It's not necessarily unique to license renewal. That's correct.

MEMBER STETKAR: And Dave, the vaults are generally --

MR. WERKHEISER: Yes. Dave Werkheiser. I'm the Senior Resident at Beaver Valley Power Station. I actually crawl down into these vaults in question, so I am qualified to go down there and these are approximately 17 to 20 some feet in depth, so after at about 5 feet they tend to be isothermal. We have not seen issues with them freezing or any issues that manifest themselves at a plant.

MEMBER SHACK: Are they covered?

MR. WERKHEISER: Yes, they are covered.

MR. HOWARD: We actually jumped ahead a little bit. We'll go back to slide 9.

First off, are there any more questions on the inaccessible medium-voltage cable issue?

The Boral Surveillance Program for Unit 1 was a new aging management program that was submitted to the staff after the SER with open item was issued in January The program was evaluated by the staff and it was determined that aging would be adequately managed by 10 CFR 54.21(a)(3).

I'd like to point out that this program is only applicable to Unit 1, because Unit 2 uses Boroflex and Boroflex is a neutron-absorber.

The containment liner issue. On April 23, 2009, during a scheduled Unit 1 IWE inspection, a paint blister was discovered on the containment liner, revealing through-wall corrosion. The staff issued RAI B.2.3-4 on May 7, 2009 requesting the applicant explain how the recent plant-specific operating experience would be incorporated into the IWE AMP.

Next slide.

The actions taken to address this issue for both Units 2 and Unit 1 on the next outage, they'll do 100 percent visual exam of the liner plate. They're also going to UT to repair area on Unit 1 during the next outage. For each subsequent outage on Units 2 and Unit 1, they're going to resume their regularly scheduled IWE visual examinations of the liner plate. And the last two items are the commitments, 32 and 33, where they are committed to do volumetric exams of 75, one foot by one foot areas of the liner plate to ensure 95 percent confidence level. Those are commitments.

MEMBER POWERS: Ninety-five confidence on what?

MR. HOWARD: That was from the acceptance criteria in the IWE regulations.

MR. ASHAR: I am Hansraj Ashar from Division of License Renewal. I will try to address what the questions you might have about the level of confidence. You spoke to the acceptance criteria when they do the UT, it would be according to the IWE requirement which allows ten percent of liner degradation without any action to be taken. If it's more than ten percent, they are to perform repair installation and show that the liner integrity is maintained.

VICE CHAIR ABDEL-KHALIK: But the question is 95 percent confidence level. What does that mean?

MR. ASHAR: It means that your chances of not detecting something in the degradation area would be 5 out of 100. That is what it means.

VICE CHAIR ABDEL-KHALIK: What could replace the chance that you're more than 10 percent, your 95 percent sure you would have found that.

MR. ASHAR: Let me explain a little more. What they are doing is they are going to perform UTs. UTs cannot accept this kind of a through-wall. Where are they going to do it? They are going to do the areas which are suspect areas, okay, that means where they find some kind of flaking of a coating or where they find some bulging of the liner plate and so many areas out that they are going to go through around the entire containment and make sure that they cover all the areas which are it may requires more than 75. If they find so many places where they're to do UT. But they are right now committing to 75 samples.

Now if they find more of them, they ought to expand their base. That is part of the requirement, this particular requirement, they increase their sample size.

MEMBER POWERS: As I understand, there will be -- and I don't understand it very well, you're going to do the 75 one foot by one foot areas and you're going to be 95 percent confident that there is no corrosion deeper than 10 percent in those 75 square feet of area.

MEMBER SHACK: If they find no corrosion and no 75 areas, then they're confident.

MEMBER POWERS: That is a kriging analysis.

MEMBER SHACK: Okay.

MEMBER POWERS: How did you do the kriging analysis?

MS. BRADY: This is Bennett Brady, Division of License Renewal. My understanding of it is if they do the sample and they get an estimate of how many flaws it is, they will be 95 percent certain that it is 95 percent free of flaws. That's my interpretation of it.

MEMBER POWERS: There will be 95 percent confidence that there are no flaws deeper than 10 percent or the entire surface area.

MS. BRADY: Not free.

MEMBER POWERS: How did you do the kriging analysis?

MEMBER ARMIJO: Some of us don't know what that analysis is.

MEMBER POWERS: Well, you've got a big area, you sample pieces of that area and that tells you something about the whole area. Okay?

MEMBER ARMIJO: But does anybody do that?

MEMBER POWERS: They did it. They took it out of FDTR 7514, Chapter 4.

MEMBER BLEY: The letter we mentioned we received that a declaration attached to it, but as I recall argue that these should be wholly random samples rather than looking at the vulnerable areas. What's the staff's -- have you thought that through. Do you have a position on that?

MR. ASHAR: Yes sir. Let me give a little historical background on this table, the degradation of the containment liner. Before this Boral instance we had four containments which had been subjected to this type of degradation. One was -- I start with North Anna, Summer, then Brunswick and D.C. Cook. In case of North Anna, it was the same reason: two by four liner between the liner and the concrete and acidity prevailed and it started corroding from inside and with the time, that was a long time, about 15 to 20 years after the log was put in probably, accidentally or inadvertently. It appeared to be start corroding in and in and in.

Now the evidence didn't show before that because they do regular examinations. If it was just shown when they showed the rust, removal of the rust coming off much larger than what can contain and then it starts showing the coating degradation. So in case of North Anna, it was two by four log. It case of Surrey, it was same two by four log, but it was in the dome area. The dome, top of the dome, the concrete area is the pressure-retaining boundary. So there was no problem so they corrected everything and they saw the dome generally is good enough.

In case of Brunswick, they found two through holes during a routine examination. The rust coming out. And then when they did the UT and they scour out the area just like what we already did, they found out that they're in one particular hole it was, I believe, a worker's glove stuck between the liner and the concrete and was creating acidity and that made it -- what they did after that, after that they went through a number of areas just like some degraded areas which will tell you hey, these are the areas of suspect that it might have something going on there, either due to bulging or buckling or liner code integration. Where will they see the venting problem UT? But to make sure that it is not same type of instance is not going on anywhere. And they continued to do that during the subsequent inspections. They did not do any kind of sampling or anything. They said wherever we see this, we are going to do this UT.

MEMBER BLEY: Thank you.

MR. HOLIAN: This is Brian Holian. Just to summarize, Hans just gave you some background and operating experience that he's looked at as part of the staff's review for this. One, it not being a new issue. We have see instances where foreign material inside the concrete have caused similar type degradation in the liner. So that was why he was bringing that up.

The staff still owes you a response in the 95 percent probability. As I mentioned in the introduction we have that in the letter that just came in yesterday from Citizen Power on Beaver Valley and we will clarify that sampling in the 95 percent confidence.

MEMBER BLEY: Thank you.

VICE CHAIR ABDEL-KHALIK: I'm still trying to understand in words what that means. Let me try something. If I do these 75 -- if I test these 75 one foot by one foot location and find that none of them has more than 10 percent loss, then I'm 95 percent confidant that the entire area will not have more than 10 percent loss.

Now if that is the case what if we find that some of these 75 samples have more than 10 percent loss? What would be the meaning of this testing process?

MR. HOLIAN: If there's more, we would expect that they would expand their sample size, expand the sample size and follow the guidance in that EPRI document. And we'll summarize that for the staff in our response to that letter.

MEMBER BLEY: And that is part of that commitment.

VICE CHAIR ABDEL-KHALIK: Thank you.

MEMBER ARMIJO: I have just one question that I meant to ask the applicant. Is it the staff's opinion and the applicant's opinion that the water that was ultimately causing this problem is a continuing leakage somewhere between the liner and the concrete or just an early live leakage retained in the wood and somehow over time maintain the right moisture corrosion conditions to cause this localized failure? What is the staff's position? Do you believe there's active leakage or not?

MR. BELLAMY: Based on recent inspections that regional specialists have done, the staff has concluded and this conclusion is in writing in the inspection report that was just issued this week, is that that water is localized and it is from the wood that was embedded between the liner and the --

MEMBER ARMIJO: Somewhere early in life or during construction this wood was soaked with water and retained it and kept that --

MEMBER BROWN: It was in the concrete.

MEMBER ARMIJO: That was the source of the water. It's not active leakage from some other location.

MR. BELLAMY: That's correct.

MEMBER ARMIJO: Okay.

MR. HOLIAN: This is Brian Holian again and that's probably what the operating experience at home has brought up. There's been other cases where a piece of wood, two by four, whatever, has gotten into a construction phase and has exhibited itself years later in this type of behavior in the liner. I will bring up though that the July 7th letter that we just received yesterday from Citizen Power does question that root cause and they question whether subatmospheric containments in general and Beaver Valley being one of those does -- questions, whether there's a mechanism that also will draw water into that liner concrete aspect in some methods. So the staff will address that, as I mention, when we address that letter.

MEMBER RAY: You've got to assume that concrete has cracks in it. In that regard, the inspection report you just spoke of references the analysis that the applicant talked about in terms of leak rate and then he has the following statement. "Accordingly, the licensee determined that the estimated containment leakage rate was within the maximum allowable leakage rate specified in the technical specification."

That clearly is taking credit for the leak- prevention function being performed in the -- what's referred to here as the other nuclear facility by the concrete. And yet, everybody stipulates that well, no, we're not supposed to do that. I don't understand how you reach this conclusion about the containment leak rate wasn't exceeded by a hole in the liner that is as big as this one was. If the basis for that conclusion is simply to say well, somebody else tested something similarly, we scaled it up, and it was within the allowable leak rate because, as I say, that's taking credit for the concrete in terms of what leakage is measured.

Can somebody, Brian or somebody, speak to that? Is that the position?

MR. ASHAR: This is Hans Ashar, Division of License Renewal. Let me explain the pumping pressure during the ILRT, integrated leak-rate testing is close to about 45 peak calculated pressure. Peak calculated pressure is much lower than the containment design pressure. At that time structurally the concrete, as well as the liner stays together and they are mostly in the elastic range. There is not much cracking in concrete, so what is happening that when you pump up to 45 psi or 39 psi, what is the peak calculated pressure for that particular plant, what happens is that concrete helps in retaining leakage at that time because the concrete is in good shape outside the liner.

Although the liner is giving away, the concrete is still resisting it. That's why at least in three cases I remember they performed ILRT just two years before they found this particular area. Instead, it met the requirement of the specifications.

MEMBER RAY: You just said a statement which is that the specifications can be met taking credit for both the liner and the concrete.

MR. ASHAR: And the concrete.

MEMBER RAY: Okay, if that's your

position, I think we want to think about that. I'm very surprised, to put it mildly.

MR. ASHAR: I'm not saying because of any theoretical reason, this is what we have found. In case of the railing. They have done the ILRT just in 2006.

MEMBER RAY: I know, but you're mixing up what is measured in an IRLT with what the function of the structures is and design basis. And I just think you want to think about that some more.

MR. HOLIAN: I think we understand the question. The applicant also tried to respond, I think, with the aspect of they respond on the visual examination of the liner during the ten-year period to also verify. So I think what you have the staff and the applicant stating is we do use this gross measure as a confidence piece, but that it's the IWE visual examination that they credit. Now they're going to supplement it with UT.

MEMBER BROWN: Well, that's fine, Brian. I'm just saying that the conclusion as stated in the inspection report here, I don't think it's correct, because I would have said that the hole as Dana I think was trying to say was a big hole and it would have exceeded the tech spec limit, but for the fact that you're taking credit in this analysis here I just referred to the concrete structure. If that's what you want to do I just think you need to think about that very carefully because I don't believe it's --

MEMBER MAYNARD: I don't agree, Harold. I think we've got to be careful that we don't penalize conservatism and design and design assumptions and stuff. The real intent of the containment system is to retain the fission product. I believe that to try to translate a small hole in the liner as saying okay, that's going to go directly out to the atmosphere and you have to be able to meet your overall design, Part 100 requirements and stuff without taking credit for your entire containment system, I think we're penalizing some of the conservative -- we do this in a lot of cases where we say all right, it's really the entire system that we're counting on, but we're going to go ahead and assume that all the rest of this is not there, but in reality it is there. I think we have to be a little careful.

MEMBER BLEY: I think we've got it on the record here. The rebar is still there, Harold, and the concrete is still here.

MEMBER BROWN: The IRLT does not subject the containment concrete to the stresses that it's designed to withstand. In other words what I'm saying to you is that the IRLT tests the membrane integrity, that's fine. But the leak rate has got to assume a design basis event which doesn't occur during an ILRT. And therefore, when you find a hole in the containment liner, you have to ask yourself do I want to take credit for the concrete during a design-basis event or containment integrity? Yes or no. That's all I'm saying.

MEMBER BLEY: The strength of the containment comes from the rebar, not the concrete. We need to go forward because even given our late start time, we're approaching the end.

MEMBER SHACK: The screaming among the members which we can resolve later.

VICE CHAIR ABDEL-KHALIK: Just a point of information. Of the four plants that you mentioned as having containment liner issues, is any of them a sub-atmospheric container?

MR. ASHAR: No.

VICE CHAIR ABDEL-KHALIK: None.

MR. ASHAR: Brunswick was BWR.

VICE CHAIR ABDEL-KHALIK: Thank you.

MR. HOLIAN: North Anna and Surrey are sub-atmospheric, but I don't know if he had operating experience.

MR. ASHAR: Surrey did not have a liner problem.

MR. HOWARD: We're going back to Dr. Bellamy right now.

MEMBER BLEY: And then we need to do that kind of quickly.

MR. BELLAMY: The only other comment I would make is that in addition to the medium voltage cables, we did have a specialist on site that took a look at the containment liner issue. Mr. Werkheiser and I made a number of containment entries. We observed the liner penetration. We observed the piece of wood, the repair activities were physically observed by the region. That's all documented in the inspection report that has been issued. And the bottom line conclusion in the inspection report is that there were no findings identified by the NRC with respect to the licensee's identification, evaluation or correction and implementation of a repair program for the containment liner presentation. That's all I have.

MEMBER BLEY: thank you very much. Anything else from my colleagues?

Well, I'd like to thank the staff and

First Energy for very good presentations and a good discussion.

Mr. Chairman, we're almost on time, given our late start.

(Laughter.)

CHAIR BONACA: Any other questions from members on this topic? If there are no questions, we will take a break for 15 minutes and restart again at 10:30.

(Off the record.)

CHAIR BONACA: Let's get back into the meeting and the next item on the agenda is the Draft Final Regulatory Guide 1.215, Guidance for ITAAC Closure under 10 CFR Part 52.

VICE CHAIR ABDEL-KHALIK: No. We're on 1.100 right now.

CHAIR BONACA: This is the modified.

(Off the record comments.)

CHAIR BONACA: We have been changing it around. Okay. So that's Draft Final Revision 3 to Regulatory Guide 1.100, Seismic Qualification and Mr. Stetkar will take us through it.

MEMBER STETKAR: Thank you, Mr. Chairman.

The purpose of today's presentation is to brief the Committee on the Draft Final Regulatory

Guide 1.100, Revision 3, entitled "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants." Proposed Revision 3 of this regulatory guide was issued for public comment as Draft Guide 1.175 in May of 2008. The comment period closed in July of 2008.

Today we'll hear presentation from the Staff regarding fundamental elements of the regulatory guide and I assume a summary of the public comments and the resolution of the public comments.

Mr. John Burke will be presenting the material from the Staff and assisting us today the Committee is Mr. P.T. Kuo who's one of our consultants and has been very active in the area of seismic qualification material.

I am not aware of any requests at this time for comments by members of the public or other stakeholders. I understand that we do have people on an open bridge line. That bridge line has been put in the listen only mode so you're capable of hearing what we say, but we're no anticipating comments by anyone at this time.

With that, I would like to turn it over to Mr. Stu Richards of the Staff who I understand would like to make some comments.

MR. RICHARDS: I think you did an excellent --

MEMBER APOSTOLAKIS: Mr. Kuo has changed.

MEMBER SIEBER: Yes. Can't you tell?

MEMBER APOSTOLAKIS: When?

MR. KUO: Excuse me. I retired last March and then I think ACRS strong searched my consultant.

MEMBER APOSTOLAKIS: You had a wonderful experiences sitting over there.

(Laughter.)

MR. KUO: Thank you.

MEMBER STETKAR: And with that, Mr. Richards.

MR. RICHARDS: I don't think I can add much to your introductions, but thank you very much. We're glad to be here and John Burke's going to lead our discussion. He's supported by Goutam Bagchi and Ching Ng from NRO and because this is a multi-discipline reg guide there's a variety of staff in the audience that are all here to support the discussion. Unless there's any questions.

MEMBER STETKAR: I did want to ask a question that wasn't quite clear to me. Are you requesting a letter from the Committee regarding the reg guide? When we originally set up this meeting, it was primarily an information only question and answer type topic. Are you asking for a letter from us?

MR. RICHARDS: Yes, we are.

MEMBER STETKAR: Okay. Thank you.

MR. RICHARDS: All right. John.

MR. BURKE: All right. Good morning. I'm John Burke from the Office of Research and what we're going to talk about today is the Reg. Guide 1.100 Revision 3 and we'll go over the background, some of the significant changes from Revision 2 and then some of the public comments and how we've resolved those public comments.

In all electrical and active mechanical equipment important to safety it must be seismically qualified in accordance with 10 CFR 50 Appendix A, GDC 2 and 10 CFR 50 Appendix B, Criterion III. Two standards have been provided or prepared by industry to provide methods on meeting the seismic qualification requirement and that's IEEE 344. 2004 is the latest version of that and it's the recommended practice for seismic qualification of Class 1E electrical equipment and then there's an ASME standard QME-1-2007 for mechanical equipment.

MEMBER APOSTOLAKIS: To what extent do

these overlap?

MR. BURKE: The ASME standard for mechanical equipment refers back to the IEEE standard some for seismic qualification. But the ASME standard also addresses functional qualification in addition to seismic.

MEMBER APOSTOLAKIS: Can you explain the difference?

MR. BURKE: The functional qualification is more in valves where it's demonstrating that the valve will stroke or remain functional.

MEMBER APOSTOLAKIS: Under a seismic event.

MR. BURKE: During and after a seismic event.

MEMBER SIEBER: With and without.

MEMBER APOSTOLAKIS: Whereas IEEE does what?

MR. BURKE: IEEE does the same for electrical and I&C equipment.

MEMBER BROWN: I was going to say it has functional in it.

MEMBER APOSTOLAKIS: Yes.

MEMBER BROWN: I got the impression from your statement that the IEEE had no function. It was just whether it broke or not, fell apart. And I was waiting for --

(Simultaneous conversations.)

It's just that the functional part -- My understanding was that initially you're incorporating -- I've forgotten how when I read the stuff. There were separate documents.

MR. BURKE: Yes.

MEMBER BROWN: And now you're just kind of moving things together with reg guide saying, "Hey, here's this one and this one that are going to deal with both the functional as well as the mechanical and both electrical and then valves and other mechanical stuff." Is that --

MEMBER APOSTOLAKIS: Class 1 includes mechanical.

MR. BURKE: That was electrical.

MR. MANOLY: This is Kamal Manoly from NRR. The previous revision to the reg guide had only endorsed IEEE 344. This is the first time we're endorsing one for electrical and another one for mechanical.

MEMBER APOSTOLAKIS: Does the IEEE standard include mechanical equipment?

MEMBER SIEBER: No.

MEMBER APOSTOLAKIS: It does not.

MEMBER SIEBER: No.

MEMBER APOSTOLAKIS: Okay. So then we're --

MR. MANOLY: That is the major change.

MR. BURKE: And we're going to get into that more.

MEMBER APOSTOLAKIS: Sometimes the answers are very simple.

MR. BURKE: Yes.

(Laughter.)

MEMBER APOSTOLAKIS: Even when I ask the question.

MR. BURKE: All right. A little history. The last revision of this reg guide was in June 1988 and that Revision 2 described methods acceptable for the seismic qualification of electrical and mechanical equipment and endorsed IEEE 344-1987 which was at that time the latest addition of IEEE 344.

There was a separate reg guide, Reg Guide 1.148, for functional specifications for active valve assemblies and that is Rev 0 March 1981 and it was methods the Staff considered acceptable for functional qualification of active mechanical equipment. That reg guide endorsed ANSI Standard N278.1-1975. Well, that ANSI standard has since been replaced by the ASME QME standard that we're now reviewing and endorsing.

In 2007 we began the process of revising this reg guide and the draft, Draft 1175, endorses IEEE 344-2004 and the ASME QME standard and like I said previously this is the first time we're endorsing the ASME QME standard.

Pardon?

MEMBER APOSTOLAKIS: Do you find the IEEE standard useful?

MR. BURKE: Yes. It's --

MEMBER APOSTOLAKIS: You kind of hesitated. Is it high level? How many other IEEE standards does it cite?

MR. BURKE: Not many.

MEMBER APOSTOLAKIS: Well, that's surprising.

MR. BURKE: This is the only IEEE standard specifically for seismic qualification of equipment.

MEMBER APOSTOLAKIS: And is it specific enough you think? I mean it's an unusual standard.

MR. BURKE: It's specific.

MEMBER APOSTOLAKIS: Okay.

MR. BURKE: And very detailed in some areas.

MR. BAGCHI: It has a long history. It is very useful. It has criteria for the excitation time history how they need to be developed and all of those things are incorporated.

MEMBER APOSTOLAKIS: In my experience with I&C --

MEMBER POWERS: It's not good.

MEMBER APOSTOLAKIS: -- have not been very good. Your silence is telling.

MEMBER BROWN: They are very high level.

MEMBER APOSTOLAKIS: And they are secular.

MEMBER BROWN: Yes.

MEMBER APOSTOLAKIS: Please.

MR. BURKE: All right. So some of the differences --

(Off the record comments.)

Some of the major differences between Revision 2 and Revision 3 of this reg guide is that Revision 3 encompasses both seismic qualification of electrical and active mechanical equipment and functional qualification of mechanical equipment and this revision expands the guidance on using earthquake experience-based methods for seismic qualification. A guidance was added for qualification and high frequency sensitive equipment and this reg guide incorporates input from NRR, NRO and NMSS.

MEMBER SHACK: Just curious. Why did you include the long history on pages seven and eight in the reg guide? That really has nothing to do with the guidance to the user.

MR. BURKE: I believe that's the history on mechanical and --

MEMBER APOSTOLAKIS: For completeness maybe.

MR. BURKE: I think just one reason was I mentioned earlier Reg Guide 1.148 is the existing reg guide for mechanical and we're incorporating it into this one and we just wanted to explain that process. I would envision the next revision of this reg guide would probably cut a lot of that out.

MEMBER APOSTOLAKIS: Did it bother you, Bill?

MEMBER SHACK: It just seemed a little strange in a reg guide which is normally telling me do this, do that sort of thing to then sort of come into a kind of dispersive discussion of the history of MOV testing and some certain amount of chest-thumping in here.

MEMBER SIEBER: It will come out as a novel.

MEMBER SHACK: But that's okay. Just curious.

MEMBER APOSTOLAKIS: Okay.

MEMBER BROWN: I actually found it useful because it wasn't -- I was a little bit surprised that the last time anybody had done anything with mechanical stuff was 27 years ago and whereas even in the IEEE standard had been updated five years ago which was also a long time based on what's been learned over the years.

MR. BURKE: And this is just all in one slide how we got to or what we're doing with this revision. We have the 1988 version of 1.100, the 1981 revision of 1.148 and what those different reg guides addressed and now we're combining them both into Revision 3.

All right.

MEMBER APOSTOLAKIS: I still find it confusing when you have two boxes that say seismic qualification and functional qualification. They're both seismic.

MEMBER STETKAR: No. Functional qualification is not seismic qualification.

MEMBER APOSTOLAKIS: That's what the gentleman said. Before and during and after the seismic event.

MEMBER STETKAR: And during any other conditions in the plant. This is functional qualification of a piece. It applies to valves. It also applies to pumps and non-metallic parts as mechanical.

MEMBER APOSTOLAKIS: Dependent of the earthquake or including the earthquake and other things.

MEMBER SIEBER: Yes.

MEMBER STETKAR: It is not related to earthquake qualification.

MEMBER SIEBER: Right. It's before the earthquake.

MEMBER STETKAR: It's functional qualification. There's a stroke time from motor-operated valve, for example, to isolate some system. That stroke time must be maintained under any plant operating conditions.

MEMBER APOSTOLAKIS: So the box that says "seismic qualification" includes functional qualification during earthquakes. That's what it means not the other way around.

MEMBER STETKAR: The other way around.

MEMBER SIEBER: The other way around.

MEMBER APOSTOLAKIS: The other way around. What does the box "seismic qualification" include? Let's start with a simple sentence. What is it? There is an earthquake and I want to make sure of what?

MR. BURKE: The equipment continues to perform its safety function during and after the earthquake.

MEMBER APOSTOLAKIS: And that's not called seismic functional qualification. Could it be called that?

MR. BURKE: Yes

MEMBER SIEBER: It could.

MEMBER APOSTOLAKIS: It could. And on the right then is not known seismic functional qualification.

MR. BURKE: That's right.

MEMBER APOSTOLAKIS: Now it's clear. Based on the previous answer, it was not clear.

MR. BURKE: All right.

MEMBER APOSTOLAKIS: Good.

(Off the record comments.)

MEMBER STETKAR: Do you know by the way why did ASME bundle together the seismic and the functional qualification in a single standard?

MR. BURKE: I don't know.

MEMBER STETKAR: Okay. Continue. Ignore the side call conversations.

MR. BURKE: Okay. Again with the major changes in this revision, combining the IEEE standard and the ASME standard into one regulation or one reg guide, all of the guidance for seismic or regulatory guidance for seismic qualification is in one document now instead of two separate documents. And the regulatory efficiency would be improved and the consistency would be improved by having everything in one document. And as we get further into this presentation you'll see there were several comments from the public relate to this.

So this revision, like I said, endorses ASME QME 1-2007 which has a lot of lessons learned for operating experience of active mechanical equipment. The existing Reg Guide 1.148 will be withdrawn when this revision is approved.

One of the biggest changes in this reg guide is related to use of earthquake experience-based methods. In the last revision in 1988, there was one sentence in that reg guide that addressed of use of earthquake experience data and it basically said if you're going to use earthquake experience data you have to submit it for the staff review and approval.

That's been expanded greatly in this revision. And both the IEEE standard and the ASME standard have extensive discussions in their standard about use of experience database and the reg guide has greatly expanded the regulatory positions related to that. So the Staff finds that experience-based methods would be acceptable if you can demonstrate similarity with a seismic excitation and a physical and dynamic characteristics between the item you're attempting to qualify and the items in the database.

MEMBER APOSTOLAKIS: Now I would like to understand that a little better. What does that mean? Can you give me an example? A simple example?

MR. BURKE: I'll try. A seismic experience database has two pieces to it, but the one that's probably the easiest to explain is going back to USI A-46 and SQUG if you're familiar with that terminology back 20, 25, years ago.

MEMBER BROWN: What's a SQUG again?

MR. BURKE: SQUG is Seismic Qualification Utility Group and this was the older --

MEMBER BROWN: SQUE or SQUG?

MR. BURKE: SQUG.

MEMBER SIEBER: SQUG.

MR. BURKE: This goes back to the older plants. The original IEEE 344 standard was 1971. There are quite a few of the existing plants where their licensing basis predates that IEEE standard. So the seismic qualification of equipment was handled a lot different back then.

So USI-A46 addressed that issue how do we qualify equipment that is older than the IEEE standards. One method used was the use of earthquake experience data. In that you have, say, oil refineries, fossil plants, industrial facilities that have experienced real earthquakes and industry went to those facilities to see what survived and what did not survive that actual earthquake and then characterized it to what was the strength of that earthquake, what was the ground motion as best as could be determined, what were the characteristics of that equipment whether it was a circuit breaker or a relay or a valve or a pump that made it fail or led it to survive and continue to function and that database is the earthquake experience database. And that process was used to justify the plants that were under the umbrella of USI-A46.

MEMBER STETKAR: John, didn't that -- doesn't that earthquake experience database also include some results from testing?

MR. BURKE: Yes, it does.

MEMBER STETKAR: And arguments that similar groups of equipment you can demonstrate similarity to a certain type of equipment that was undergoing a test, a relay or a switch or something like that.

MR. BURKE: Right.

MEMBER APOSTOLAKIS: But the qualification for a given nuclear plant actually don't count what the design basis earthquake.

MR. BAGCHI: Yes.

MR. BURKE: Yes.

MEMBER APOSTOLAKIS: So not all experience might be relevant.

MR. BURKE: Correct. You have to -- I meant the earthquake experience database is grouped by classes. So take an electric motor. One class might be half horsepower to five horsepower or 20 horsepower to 200 horsepower. That's a class of equipment as what we're talking about here.

MEMBER APOSTOLAKIS: And now if I have a safety related component in a nuclear plant and a similar but not safety related component in a chemical plant that exhibits a certain behavior, how do I relate it?

MR. BURKE: You would -- If the equipment that you're attempting to qualify in your nuclear plant fits the similarity and the dynamic characteristics and it's the same model number or similar model number as what survived in that fossil plant and then you compare the actual seismic ground motion to your required response spectrum at the power plant and if it envelopes, then it's qualified.

MR. BAGCHI: John, is it not appropriate to point out to the ACRS that primarily what we're talking about for the change in Revision 3 here is recorded testing of past seismic shakable testing and response spectra that were recorded from that experience. So there is a base of information that already exists.

MEMBER APOSTOLAKIS: For safety-related components.

MR. BAGCHI: For any component that you want to consider for seismic qualification.

MEMBER APOSTOLAKIS: I'm more interested in safety-related.

MR. BAGCHI: It is all applicable really to safety-related components.

MEMBER APOSTOLAKIS: But in other

industries they don't have that kind of thing.

MR. BAGCHI: No, that's right.

MEMBER APOSTOLAKIS: So it would be hard it seems to me to take experience in an oil refinery.

MR. BAGCHI: Absolutely. Not only that. It would be very hard to determine actually what response spectrum that was experienced that could be attributed to the successful functioning of a piece of equipment. But it was done with the help of a panel and everything else and I think in this regulatory guide we do not endorse it for any plant other than the A46, USI A46 plants.

MEMBER APOSTOLAKIS: Thank you.

MR. BURKE: All right. Another change in this reg guide was we added guidance to the qualification of equipment sensitive to the high frequency excitation and that guidance is consistent with the interim staff guidance used for new reactors.

And as mentioned previously, the public comment period was from May to July of last year and we received 84 comments from the groups listed. A lot of the comments were similar or overlapping. Like the comments from IEEE and comments from Westinghouse may have been the same issues. And then we had a public meeting in December to address the comments and we had representatives from those groups listed.

I thought we'd discuss some of the major comments here and we've already talked about one of them which is why are we combining Reg Guide 1.100 and Reg Guide 1.148 into one document. In the draft guide that was sent out for public comment the reasoning wasn't well explained and that was one of the reasons the background section was expanded to give a better history on it and explain why we're combining them and, like I said, when this revision is approved, then we'll withdraw 1.148.

MEMBER STETKAR: John, you mentioned -- Could you elaborate on that just a little more because I quite honestly found it confusing to myself that these two different sets of qualification criteria would be bundled into a single regulatory guide given the historical separation of the two.

From the staff's perspective, there were several comments regarding the fact that it wasn't clear why they were being combined. Could you elaborate a bit more on the staff's perspective of why this is either more efficient from a regulatory perspective or less confusing to a potential user of the guidance?

MR. BAGCHI: One regulation where it all

comes together is the qualification regulation under 50.48 I believe. That's the qualification that's required for environmental conditions and it includes seismic.

PARTICIPANT: 50.49.

MR. BAGCHI: I'm sorry. 50.49. It includes seismic. That's why in the sequence of qualification to meet the regulation seismic comes last and it is appropriate to put them altogether in one kind of reg guide where seismic is the final stage of the qualification.

MEMBER STETKAR: I'm asking more along the lines of bundling the non-seismic functional qualification guidelines with the seismic qualification.

MR. BAGCHI: I don't have an easy answer to that.

MR. SCARBOROUGH: This is Tom Scarborough with NRO. This goes way back to when we were working originally on QME-1 to develop a standard that the Staff could endorse way back to Jim Richardson's day back in NRR. The way ASME had written QME-1 they included seismic with the functional qualification and that was their scope. That's how they worked it.

So as we work through over the past 20

years working closely with ASME to develop this standard that we could endorse when we got to the end, we have a seismic portion of QME-1 and we have a functional qualification, the flow testing, that sort of thing, where we incorporated like Dr. Shack said the history of MO dropper valves (phonetic) and all the lessons learned and the internal clearances and the dimensions that we found to be critical for flow testing under high flow conditions. So that's all bundled into the very specific guidance in QME-1-2007.

So we get to that point and then we have one whole standard and we have to decide do we like split this and endorse, write two reg guides and have one reg guide endorse the seismic portion and another reg guide endorse the functional qualification portion.

And what we decided was that the functional qualification was such a clean endorsement. You'll find these almost no conditions placed on functional qualification side and so since it was such a clean endorsement it was easy just to piggyback it right into the standard.

A user picking up QME-1, they can use it for everything now if they go to Reg Guide 1.100. They don't have to pick up two reg guides to use this standard and that was the reasoning. It was just a simple logistics in terms of simplicity of it. We didn't have much to say about the functional qualification because we'd spent 20 years reading the standard the way we thought would be appropriate. So that's the reason.

MEMBER STETKAR: I guess I understand that. Reading the public comments I guess I'm curious. Did you have meetings after the public comment period?

MR. BURKE: Yes, we did.

MEMBER STETKAR: How did those meetings go because a lot of the public comments seem to say they wanted to keep the two reg guides separate regardless of the fact that they both referenced the same standard?

MR. SCARBOROUGH: And that's one approach we could have taken and we discussed this with them at the public meeting. That's absolutely one way we could have taken it. At the time, we just made a decision whether to have two reg guides to endorse one standard or sort of one reg guide to endorse the standard which covers sort of two areas of review and we just thought from a efficiency point of view it was just easier just to go with the one reg guide and piggyback the functional qual because there were almost one conditions to place on that portion of the standard.

And we explained that and they sort of saw our approach and if we had to do it over again we might have done it a different way. But that was the decision we made at the time to try to move it through as quickly as possible.

MEMBER STETKAR: Okay.

MR. MANOLY: It was a -- purpose.

MEMBER STETKAR: Thank you.

MR. BURKE: So another -- The use of earthquake experience methods is addressed in both the IEEE standard and the ASME standard and this comment concerns the earthquake experience methods for ASME equipment and we mentioned briefly the USI-A46 and the SQUG and the industry requested approval to use the SQUG methodology for qualifying new equipment.

And the staff does not accept this SQUG methodology for non-A46. However, we do accept --

MEMBER BROWN: What's the difference between -- I'm not familiar with A46. Okay.

MR. BURKE: Like I said earlier, A46 is that subset of existing plants that more or less predate the early '70s. I don't know exactly the cutoff. It's a '74.

MEMBER BROWN: So for plants subsequent to that you don't accept the SQUG data.

MR. BURKE: That's correct.

MEMBER BROWN: All right. That's -- You've answered. I don't mean to get --

MR. BURKE: And it's roughly half, isn't it? About half of the existing plants.

MEMBER STETKAR: It's about 70 units.

MEMBER BROWN: Okay. I just needed to understand the basic difference there and I've got it.

MR. CHEN: This is Pei-Ying Chen from NRR.

MEMBER BROWN: Don't confuse me.

MR. CHEN: No, no. I can help you out.

MEMBER BROWN: That's very hard to do.

MR. CHEN: Yes, I used to handle USI-A46 code for maybe more than a dozen years. So I know a little bit about the history. The reason there is an A46 -- is at the time most of the new plant were ***11:03:18 1975. So it's an improved criteria in 344-1975. So all the plants which were qualified before that was put into USI-A46 plan which is about 70 some plants at the time.

And then because of the difference in qualification at the time of license the whole USI-A46 was resolved in a different sort of criteria than the later plant. So that is really the basic.

MEMBER BROWN: Okay. I got it now. You did not further confuse me. I think I'll stay on the same track. Thank you.

MEMBER STETKAR: John.

MR. BURKE: Yes.

MEMBER STETKAR: How does the reg guide -- I may have missed it in my reading. How does the reg guide treat qualification of new equipment for the A46 plants? Suppose I want to install a new digital instrumentation control system for example in my A46 plant. Can the licensee use the SQUG methodology to qualify that equipment, the cabinets, the anchorages and things for the new plant?

MR. MANOLY: Yes, this is Kamal Manoly again from NRR. All the plants that were under A46 ended up in group rating that procedure in their SARS.

MEMBER BROWN: Which procedure do you mean?

MR. MANOLY: For the Generic Implementation Procedure which implements the SQUG methodology. It became part of the SAR for all these plants for replacement equipment and modifications in the plant, for equipment that meet the criterion in the GIP. So the equipment that fit in the classes that's described in the GIP they can basically use the GIP.

MEMBER BROWN: What's the GIP?

MR. MANOLY: The Generic Implementation Procedure.

MEMBER BROWN: Say that again.

MR. MANOLY: The Generic Implementation Procedure. That was used for qualification .

MEMBER BROWN: So they don't have to meet the new standards.

MR. MANOLY: No.

MEMBER BROWN: Okay.

MR. BAGCHI: However, if it's a piece of equipment that's brand new and does not fall in that database they have to qualify.

MEMBER BROWN: Well, I'm just a little bit -- I question that I mean. That was 1975 and earlier and now they're going to put in a new set of digital I&C equipment and we're going to seismically qualify it to some experience base from pre 1975. So it's a little bit -- I understand licensing basis set before you guys leap on me. The current license, whatever they were brought under, but that just seems to be a dichotomy to me that --

MR. RICHARDS: Goutam, correct me if I'm wrong, but what I think what you just said is that if it's not in the existing category you can't do it and chances are most of the digital systems that we'd be seeing today probably didn't exist in 1975.

MEMBER BROWN: No. But there are circuit cars and boxes and I mean the crane metal cans with drawers or something come in and out. I mean they look the same, the valve and the pump and things like -- I don't know why they wouldn't look the same because those a blacksmith -- Anyway, we can go on.

MR. BURKE: All right.

MEMBER STETKAR: The main comments, on the previous slide, from the industry regarding the experience database were planted toward the use of the experience for new plants. Is that correct? Did I understand you correctly? They wanted to be able to use the experience data for --

MR. BURKE: Yes, I believe that's correct.

MEMBER STETKAR: Okay.

MR. MANOLY: This is Kamal Manoly again. I think industry would like to be able to use the approach for non A46 plants, the operating reactors. Industry has been updating the database and expanding it and we felt that they can consider it, but we need to know the database that they will be using.

MEMBER STETKAR: Yes, if I read all the qualifications in the reg guide it's pretty discouraging.

MR. MANOLY: Well, I mean I think we wanted to see a procedure that implements the code and that's the procedure that we'd like to approve first before we grant that to non A46 plants.

MR. BURKE: The existing reactors that are not A46, the 30 or so, right now they have to get approval for every application case by case. This opens that up a little bit where they can get approval for the process and not necessarily a case by case approval.

MEMBER STETKAR: I didn't think of it in that context, but okay.

MR. BURKE: There was another series of comments about the nonmandatory appendices in the QME standard and the way the draft reg guide was worded it was confusing over whether the nonmandatory appendices were now becoming mandatory by the way we were addressing them and endorsing them in this reg guide. So we clarified the language to say if your qualification program is relying on a nonmandatory appendix, then that nonmandatory appendix then becomes mandatory for you.

MEMBER BROWN: I'm obviously the slowest of the group and that's why you're looking at me.

(Laughter.)

MEMBER STETKAR: John, ignore him. He speaks up when he wants to.

MR. BURKE: Okay.

(Off the record comments.)

MR. BURKE: There was also some discussion in the draft reg guide about inadvertent high frequency content in previous tests like Goutam said earlier. Part of the earthquake experience database includes test experience, not just actual earthquakes in the field but a database of all the equipment that's been tested and how that database tested equipment can be used and what the frequency content is of those existing tests. So we clarified our regulatory positions on the use of that test experience data and how to use that.

MEMBER STETKAR: Was there much resistance to extending the frequency above 20 hertz from the industry perspective because the new requirement just says you don't accept the limitation of 20 hertz which has been historical or 33 hertz? You need to look at the site-specific response spectra.

MR. BURKE: I don't remember much about that.

MEMBER STETKAR: Okay. It's become an issue certainly in our reviews of a lot of the new reactors and specifically for the east coast siting the high frequency component. I was just curious whether you had much discussion regarding that scope of the reg guide.

MR. BURKE: I don't remember.

MEMBER STETKAR: Okay. Thanks.

MEMBER BROWN: I wanted to ask one question and this relates back to my experience in the naval program in that most of the types of testing involve -- I mean we do both shock and vibration with these shakers and stuff. So we look at the spectra and vibration tests which these have a frequency component that's routine that you have to deal with and we scan and then pick the worst and then we shake it to death at the resonant frequency or we go to the max that could be expected in the application and then test it.

In the shock test, there are three different machines like medium and barge testing, heavy weight testing, where there are specific -- You know, you blow up explosives and I didn't see how that was -- I didn't get an understanding of how that was done. I mean do they -- I mean I understand blowing up giant 60,000 pounds of HBX in some quarry somewhere and you can watch it rock and roll and get all kinds of interesting --

MR. BAGCHI: I think the heart of your question can be answered if you look at the discussion part of it which says that we look at compatibility of the power spectral density.

MEMBER BROWN: Of the what spectral density?

MR. BAGCHI: Power spectral density.

MEMBER BROWN: Okay. So you do make a judgment --

MR. BAGCHI: Yes, sir.

MEMBER BROWN: -- based on what's expected in that region from --

MR. BAGCHI: Yes.

MEMBER BROWN: -- earthquakes and then you test it. Okay. At that site. Yes. All right. Thank you.

MR. BURKE: All right. And this is some more discussion on the high frequency sensitive equipment and the interim staff guidance used for new reactors. Again, there was the draft reg guide wording was confusing in some aspects. So we tried to clarify that section about the high frequency sensitive equipment to be consistent with the Interim Staff Guidance. If you have a high frequency site, what we're doing now is we're saying if the high frequency sensitive equipment needs to be qualified consistent with the interim staff guidance both for new reactors and existing reactors.

And there were a lot of comments about the use of the test experience data being too restrictive like we mentioned a little bit earlier. Quite a bit of this reg guide the regulatory positions address use of experience data. I believe it's like 20 pages and about 12 pages are regulatory positions on use of experience data.

One of the issues and if you have high frequency sensitive equipment is the spacing of your sampling, a one-third octave or one-sixth octave. So if you have high frequency sensitive equipment then you need to test or collect data at one-sixth octave spacing where the standards currently would allow one-third octave.

MEMBER BROWN: So you're restricting the use of similarity. They're complaining about that.

MR. BURKE: Yes.

MEMBER BROWN: I'm not complaining. They had some objections.

MR. BURKE: Yes.

MEMBER BROWN: Okay.

MR. BURKE: And then developing the test experience spectrum in the draft guide we wanted an equipment capacity factor of 1.4 applied to the test experience spectra. Another way of looking at the 1.4 is a reduction factor to give us a little more margin and we revised our position on that and we've deleted the 1.4 factor because we did have a sound technical basis for imposing the 1.4. So Research has added that to the seismic research plan to develop that technical basis. Is the 1.4 a proper number or what is the proper factor to use in this situation?

And there were several comments about definition of Operating Basis Earthquake or OBE. 10 CFR 50 Appendix S was issued in the late '90s that changed the definition of what an OBE is and the SECY paper that's listed there gave a little more detail on that issue. In the reg guide, I guess, and in the draft reg guide that went out for public comment, we did not explain it well that use your licensing basis for OBE or the information in the SECY paper and Appendix S, whichever is appropriate for your plant.

So in the typical operating reactor, the OBE, is half of the SSE. But under the Appendix S, it could be much less than that. So we just clarified the wording that use whatever is appropriate for you plant. Use your licensing basis or if you're a new reactor use Appendix S.

Any other comments? I have a video if you want to see a seismic test.

DR. BLOHM: Yes. Please.

MEMBER STETKAR: We are ahead of schedule.

MEMBER APOSTOLAKIS: Go ahead.

MR. BURKE: It's a 30 second video.

MEMBER STETKAR: It's a real seismic event then.

MR. BURKE: It will take longer to load than to show it.

MEMBER APOSTOLAKIS: In color?

MR. BURKE: Yes, it's in color. There are two commercial --

(Off the record comments.)

MR. BURKE: There are two commercial facilities in the country that do triaxial seismic testing for the industry and this is one of them.

(Video played.)

And that's just a 10 foot by 10 foot

triaxial table. I don't even know what equipment that is. I asked them to send me a video and they did.

(Off the record comments.)

That's just a typical triaxial seismic qualification test and you'll see it lasts 30 seconds. So if you're doing a typical qualification program you would do five OBEs which would look like that or maybe that's an SSE. I can't tell and one SSE which would be double the OBE.

MEMBER BROWN: What's the OBE? Operating Basis Earthquake.

MR. BURKE: Operating Basis Earthquake.

MEMBER BROWN: I was guessing something dramatic.

MEMBER APOSTOLAKIS: Very good.

MEMBER STETKAR: Any other questions? Yes, sir.

MR. STARKE: My name is Richard Starke. I work for MPR Associates. I sat on both the ASME code committee working group that developed the revised standard as well the IEEE 344 working group.

I guess in one sense I'm a little bit disappointed in the reg guide from a major perspective. If you would slip back to slide number 11, it has four bullets on that slide. The first one says that the staff position was that the use of experience method could be subject to review and approval and then the last bullet says use of experience method for seismic qualification is subject to NRC review and approval. So we put all this effort into developing standards, but it still could well be interpreted to mean that there would be one-for-one. If you have to do a shake table test and that requirement was on there, then every shake table test that is done the NRC staff would have to review and approve.

So I'm reading this to mean that use of experience data is really something that the staff is still not comfortable with and slide 21 the first bullet makes the same point again with respect to the use of experience methods which is that review and approval is required for the comparison occurrence.

So I'm left with having served on both these co-committees spent five years developing the standard and we come out with a reg guide that has almost as many pages in it and there are exceptions and clarifications and positions the staff is taking and then when it's all said and done the staff still has to review and approve it.

I guess the main concern I have from an

industry perspective is that I just don't believe that this process of the standard and the reg guide is going to bear fruit. I think this is a tree without any fruit that the utilities are just not going to use this methodology within the standard because of (1) having some many additional restrictions placed upon the use of the standard and (2) because you still have to go back to staff and get review and approval.

MR. MANOLY: May I respond to the comment please?

MEMBER STETKAR: Sure.

MR. MANOLY: This is Kamal Manoly from NRR. We thought that the major difference between this revision and the previous revision is that Revision 2 specified on a case by case basis which means every time a licensee wants to use the experience data for one valve or one small equipment, he had to go to the staff and that seemed unreasonable to us.

What we're looking for here is a procedure, you can call it topical report, that implements the QME and IEEE and provisions in the reg guide that we would review and approve and then industry after that can do it on their own by 50.59.

MEMBER STETKAR: It's interesting when Mr.

Burke sort of gave that perspective of the acceptance of the experience data. I certainly didn't get that interpretation as I read the reg guide. I read the reg guide as requiring a case by case approval.

MR. MANOLY: That was in Reg 2. But that's not --

MEMBER STETKAR: No, the current reg guide is -- The latter.

MR. BAGCHI: But for new reactor I understand that the new reactor does not accept the use of experience database.

MR. BAGCHI: Can I address part of this?

MEMBER STETKAR: Sure.

MR. BAGCHI: When those standards, national consensus standards, were being developed this entire concern about the central and western United States high frequency motion in the earthquake was just appearing and it is a very serious concern to the staff with respect to the peak ground acceleration being asymptotic to 100 hertz where the previous practice was 32 hertz and we are talking about experience information that had been developed using the peak ground acceleration possibly around 33 hertz. And any kind of experience information one might get is probably because of high level of testing beyond that was intended.

That's why it is necessary for the staff to look at some of these things in an objective basis and that does require review particularly because of the high frequency concern.

MEMBER STETKAR: Thank you.

MEMBER BROWN: The point of the comment was industry is not going to use the operating based experience. They'll just go right to the testing.

MR. MANOLY: No, they can --

MEMBER BROWN: No. Let me finish. I'm trying to make sure I understood the comment from the MPR representative and make sure I knew what that was. It's essentially saying you can still do operating -- I mean if somebody was to come in and do the operating base you can do that, but at any time somebody wants to do it, you have to look at it. Whereas if they do the testing then they just use the process of whatever it is. The standards are there. You qualify and you're off to the races. Is that it?

MR. RICHARDS: Another option is for the industry to come in with a generic process, have the staff review that. It would have more detail.

MEMBER BROWN: Do you mean for using operating basic experience.

MR. RICHARDS: Yes. And once that more generic process is approved as long as it's within that envelope then they would be able to use that.

MEMBER BROWN: But you're pushing it back to if they want to, I don't want to use the word complain, if they want to say "We really want another option" you're saying, "Fine. Refine that option so that we can treat it in the same manner as we do the testing regime with more detail in terms of how they would do it or use it." Is that it?

MR. BAGCHI: That's correct.

MR. MANOLY: And let me clarify. In the late '90s, industry submitted a topical report that we rejected because it was not complete enough in describing the data in the experience database to be used for non A46 plants. We felt that the database was not complete enough, did not have the kind of examples we were looking for that experienced severe earthquakes.

But that was a starting point. And they can complete that information and submit it again as a topical and if we approve it, then they can use it across the board.

MEMBER BROWN: Just one observation. I had to deal with similarity all the time in the Naval program and I have to admit I normally for my equipment since they were protect equipment and reactor plant monitoring equipment I normally defaulted to test as opposed to similarity.

Now I won't say that do that every time because testing is expensive and I mean in spite of the reputation of the Naval Nuclear Program having these giant bags of gold that you just open the faucet that's not really the way it is.

MEMBER SIEBER: Stimulus.

MEMBER BROWN: So we looked at them on a case basis when we were going to do that. So I understand the concern. I just -- It seems to me there's a way to do it. I just don't trust similarity real well when I'm -- with the new stuff, buying new equipment. That's all.

MR. BAGCHI: One other factor that could be considered here is that it is only related primarily that the high frequency end of the spectrum is primarily a concern for chatter-prone equipment.

MEMBER BROWN: Say that again.

MR. BAGCHI: Chatter-prone.

MEMBER BROWN: Oh, chatter-prone. Okay.

MEMBER STETKAR: Any other comments? Questions?

(No verbal response.)

With that, I'd like to thank you very much. It was a good presentation. I think we had a good discussion and P.T.

MR. KUO: Yes, I have some comments and actually more of questions than comments really. Just for everybody's information why I'm sitting here. I spend the first half of my NRC time dealing with seismic issues and the last phase of the A46 issue actually was part of my responsibility at the time and Dr. P.Y. Chen was the lead at the time on this A46. So I have a lot of old history in my mind. It faded away a little bit, but some of the memory that I still have.

I read the reg guide and I think I can say it is well written. But I do have a couple of questions. The first question I have is about the section 1.1.1.d. It says, "The use of experience data for seismic qualification of electrical equipment is subject to review by the NRC staff such as 1,2,3,4,5." And among this I don't see a mention about the site conditions. Okay.

Like Goutam pointed out that the high frequency region, I could say that there are cases that the frequency of -- the predominant frequency of the structures would be low like in Mexico City. It's a lower frequency. And if you take that experience data to the east coast for a site on the rock certainly I would not consider that is applicable. But that is not mentioned here. So that is a site condition that I'm concerned about.

And also in some of the plants especially foreign plants that they put isolator underneath. So the response of those kinds of plants will be different, too. So you take experience data from Japan for instance. They have isolators there and I don't know if that experience data really is applicable to U.S. plants.

MR. BAGCHI: The discriminating factor here is going to be the required response spectrum. There is required response spectrum. If it is designed with isolators that's what it will show up.

MR. KUO: Well, it looks like the focus is on the high frequency. I'm also talking about low frequency. Okay. Low frequency you don't look at the accelerations. You look at the displacement, the deformation.

MR. BAGCHI: Let me suggest that most of the equipment under the scope of the certified design is designed to very demanding response spectrum, rich in low frequencies, reg guide 1.60 type of spectrum and if you look at that it will amply cover anything like the new -- anything like the Charleston event influence in the eastern sites and the ground motion response spectrum will take that into account amply and in most cases they'll have to show how the site is enveloped by the certified design spectrum. And once all of those criteria are put together the required response spectrum that are defined through the instructed response spectrum which are peak rod and so many other things then we don't really have a concern about the specific siting effect.

MR. KUO: That's okay. That's good if you don't have any concern. But I used to generate a site peak -- history myself. Okay. I can play with that and if you want to add a frequency, delete a frequency, that's pretty easy. So, yes, I have a time history. I can envelope the response spectrum. No problem.

But I can -- I don't have to create some of the input in the frequencies that I might have. And -- Well, let me stop there.

MS. KAMMERER: Hi. Anne Kammerer, Office of Research. I think there are two separate things. One is the experience in terms of the actual earthquake and the other is experience in terms of the laboratory testing or shake table testing.

In terms of the actual earthquake what needs to be done for each of these case studies for the equipment is you have to look at the actual loading upon the equipment. And so in the cases you're talking about, for example, in a base-isolated plant or in a plant with, say, the west where there's a lot of long period motion you'd look at the experience and the loading in that event.

You're absolutely right in that if you have experience in an earthquake in, say, a base-isolated structure that would not necessarily mean that that equipment can withstand high frequency loadings. You're absolutely correct about that. So that's in terms of the earthquake experience.

In terms of the shake table testing, I think that was more to get to Tom's comment in that in that case you still have to do the same thing which is look at the motion that was used in the shake table testing and compare that with the demands according to the design.

MR. KUO: I agree. And that will be my next two comments is that, yes, I agree with the stance that you mentioned here to look at similarity, to look at things that pertain to the actual and what you have. However, I'm just wondering if the staff has developed any guidance for comparing similarities and make assessment of the site conditions for the reviewers to do. Okay.

Now the applicant submitted another report and say "We are going to use the experience data and we think that this is similar to that." When the reviewer that in front of them, do they have any criteria guidance to use to say, "Yes, I agree with you. This is similar. This is not." But is that the opinion of the individual if the staff has any guidance for it? That's important to be able to implement this.

MR. BAGCHI: If we have a generic report that comes in for review that generic report is reviewed through the process that has been laid out and it has been found to be robust enough that it is no swayed by individual judgment alone. Let me rest with that.

MR. KUO: Right now we don't have it.

MR. BAGCHI: We don't need to. Every technical report that comes in we cannot write a set of criteria for that.

MR. KUO: But like we discussed before,

suppose the individual comes in with an application and applied this set of experience data right now. How are you going to make your judgment? That's my question.

MR. MANOLY: This is Kamal Manoly. I think your point is well taken and I believe the second item, the exclusion/inclusion rules, we expect considerations to be captured in that bullet that there will be exclusion and inclusion rules that would distinguish between equipment that have seen high frequency/low frequency of the thought that you're talking about.

MR. KUO: I'm just making suggestions or pointing out that there might be difficulties here for the reviewer to judge. Yes.

MEMBER STETKAR: Okay. I think we're getting into an area that's more focused on review of submittals that may be made by a particular licensee or an applicant rather than the reg guide itself. I think it was a good discussion.

MR. KUO: Okay. But I have one more comment.

MEMBER STETKAR: Okay.

MR. KUO: Rock site that is quoted in the reg guide. I'm just wondering if there is a definition for that.

CHAIR BONACA: I'm sorry. I didn't --

MR. KUO: What is the rock site? We used to have the definition. It was shear wave velocity and something like a 2500 feet per second or 3000 feet per second. Here when this reg guide says rock site, do we have a definition for that?

MR. BAGCHI: We can fall back upon other sets of criteria that are used for probabilistic seismic hazard analysis. It is generally considered that 9200 feet per second produces rock conditions for which there is no side effect. We licensed AP1000.

MR. KUO: Fine. But all I'm suggesting is that you need a definition for that.

MS. KAMMERER: Let me respond to that a little bit in that. This is Anne Kammerer, Office of Research. In terms of what the definition of a rock site is that definition comes from something which is outside of this agency to some extent in that it often comes from ground motion prediction equations which are used and those are based on empirical and other types of data that are done and used more throughout the seismic community.

And so those equations, the definition comes from those equations in that it's a separation between whether directly applicable and where you might need to do additional site response as well. So we would not define that. It comes from tools which are used in a broader seismic hazard assessment.

MEMBER STETKAR: Yes, sir.

MR. PARELLO: My name is Jim Parello. I'm the Chairman of IEEE 344. I also work for Westinghouse. I have a question in regards to slide 21.

Slide 21 deals with test experience-based qualifications and in the process here it states clearly that when using this method that your expectations are that the TES curve should be provided for review and approval and that it's based on the standard deviation and mean-type data. The question is if we're qualifying a methodology, if we're going to go through that process, then this data would not be available until you perform the act of going through and generating your grouping of equipment for your test experience-based.

Are we talking about a technique or are we talking about data here? So just a point of clarification I'm looking for.

MR. BAGCHI: We're looking for data primarily. Otherwise how would we know that the test experience spectrum would go to the class.

MR. PARELLO: What we talked about earlier was that you were looking for a methodology to be approved and then we process through that qualification methodology using the data.

MR. MANOLY: I think-- This is Kamal Manoly again. In a topical report we expect to see the data that industry is using to lump equipment in classes. We want to see the earthquakes, for example, in the earthquake database. You want to -- The reference -- That's why the earthquake that's being identified how it was measured, the equipment that has seen that, how it was measure, the equipment that has seen that, does it meet certain geometrical limitations, dynamic characteristics. All that part is really part of the exercise.

MR. PARELLO: The IEEE 344 standard presently gives criteria for those types of things.

MR. MANOLY: But when you --

MR. PARELLO: When we're going through the process here, what you're saying is every time I generate a group or an actual class you do want to review and approve that specific class.

MR. MANOLY: You have -- I mean we know for A46 it has the class of 20 and in establishing the class you identify certain earthquakes, certain equipment that fell in that class, the limitations on this equipment. So that's part of the body of the data that qualifies that class. So if you're going to be establishing new classes or adding or expanding that, we'd like to see what's in the database.

MR. PARELLO: The test experience-based is dramatically different in an A46 method.

MR. MANOLY: I understand that. Yes.

MR. PARELLO: In this method, you're already using qualified test data for equipment that was seismically qualified, safety-related equipment in the process of generating this particular class. What we have here is a potential dissimilarity in any one of these test programs to the candidate equipment. The candidate equipment is the equipment that you want to qualify versus the class which is made up of similar equipment.

So we've gone through this process. You have the same qualified equipment and when you do so, you're generating a program or qualification level. Now what you're saying is you still want to see all the data to qualify it. Is that correct?

MR. MANOLY: Well, you need to identify what is the class made of. I mean, you have to reference the specific test that you're utilizing to qualify that class.

MR. PARELLO: That's not a methodology. That's the actual implementation just like if I went out and performed a qualification test instead of going through this. I mean it's the same thing. I mean, do you want to see that test data if I did a one-to-one qualification?

MR. MANOLY: I understand your question. I think that the level of the review detail would vary depending on how reliable the information is. I mean if you're talking about experience database for seismic that's a lot more complicated than just a test that was done based on determined procedures and the QA procedures. Whereas, talking about data extracted from earthquakes sites, that leaves a lot of gray areas. That's why we want to see it. So I think the distinction between the test data versus seismic experience data.

CHAIR BONACA: Can I intervene here for a second? We have to kind of keep on schedule and it's a good interchange that's going on, but it has a lot to do between staff and the industry in terms of what's required to approve a particular topical report and I think that's quite a bit level of detail below our deliberations regarding publication of this reg guide.

I'd encourage you to keep the discussion going but perhaps in a different venue from this meeting if it's possible.

MR. CHEN: Okay. This is P.Y. Chen.

MEMBER STETKAR: Or not.

(Laughter.)

MR. CHEN: I guess really what Jim is after is that he's talking about this TES, you know, how do you determine it. And basically the industry guide right now is they are talking all this spectra and then take a frequency-by-frequency mean of the data and I think the question here is that we thought that mean may not represent a good determination of the final TES level.

So what we are asking instead of 1.4 factor actually it comes from the mean plus some of the sigma that people use, actually Kennedy used. But anyway we decided not to use 1.4 factor, but we asked for the mean and the standard deviation and to decide what kind of level will be determined.

MEMBER STETKAR: Understand. Thank you. Any other comments?

MR. PARELLO: This is Jim Parello again.

My goal is to update the standard and put the appropriate language in there in regards to areas that need to be improved and that's why I'm asking the question.

MEMBER STETKAR: Understand. I understand.

MR. PARELLO: Thank you.

MEMBER STETKAR: Thank you. With that, I think I will turn it back to you, Mr. Chairman.

CHAIR BONACA: Okay. Thank you for the presentation and we will take a recess for lunch. Get back at 12:45 p.m. Off the record.

(Whereupon, at 11:47 a.m., the above-entitled matter recessed to reconvene at 12:45 p.m. the same day.)

CHAIR BONACA: We're back into session. We have -- the next item on the agenda is Applicability of TRACE Code to Evaluate New Light Water Reactor Designs, and Professor Sanjoy Banerjee will lead us through that presentation.

MEMBER BANERJEE: Okay, thank you, Mr. Chairman. I'd like to make a few remarks regarding the TRACE code which we reviewed for applicability to the EDBWR, specifically on February 27th, 2009. Now, I notice the agenda item here is a little wider which is to evaluate light water reactor designs applicability to. So I think the subcommittee did not really consider this rather broader commission, if you like. So most of the discussion was related to the ESBWR.

So probably today this is what the presentation will be all about, for in our discussion we put -- discussed, of course, whatever issue we want to and hopefully we will, and this could include what we should do about looking at TRACE for other light water reactor designs, the new light water reactor designs. So this subcommittee meeting was held February 27th, which is quite a long time ago.

At the subcommittee meeting we had ISL make an extensive presentation of work they had done under contract to the NRC on evaluating the applicability of the TRACE code. This was a pretty substantial study by all standards and there were several questions that came up. And there were also -- there was also an internal review of this report that ISL had made by NRC staff which came as an appendix to the report and this review is even more interesting than the report itself actually. So that's the basis on which we've been considering this matter, this ISL report, and with that, what I'll do is I'll turn it over to Chris Hoxie from Research to introduce the various people.

The only thing I can say is we're going to be taxed for time looking at the agenda, so I'll try to keep you moving. Okay, thanks.

MR. HOXIE: Thank you. My name is Chris Hoxie. Let me just quickly introduce the speakers we have. Dr. Ralph Landry from the Office of New Reactors is going to give us a little bit of background on the regulatory application of computer code. After his remarks, Dr. Joe Staudenmeier will be talking about sort of an overview or introduction to the use of TRACE and TRACE applicability to ESBWR LOCA and these parts we wanted to do in open session.

Joe has then split out the proprietary portion of his stuff and we will move into closed session for Joe to go over the proprietary information with you and that's followed then -- there was an interest in the treatment of the momentum equation and so Dr. Steve Bajorek is here to talk about the momentum equation, and we'll of course, take questions as they go. So with that, I'm going to turn it over to Ralph and we'll move right along.

CHAIR BONACA: Thanks.

MR. LANDRY: Thanks, Chris. My name is

Ralph Landry from the Office of New Reactors. And I asked to make a couple remarks at the beginning of the discussion today because the topic really does deal with the new reactors and our use of the code within the Office of New Reactors. So I wanted to make a couple generic application remarks first and then a couple specific remarks on how we're using TRACE in NRO and its applicability.

Okay, the purpose of codes in general in regulatory space, I think a number of you have heard me say this stuff before. It has three points. One, we want to do confirmatory analyses. We want to do analysis that give us a warm fuzzy feeling about the material that's been submitted to us by either a licensee or an applicant. We want to know are the analyses we're seeing reasonable. We want to know, have the analyses captured the important phenomena in the space of a large break LOCA.

Typically, there are 30 to 35 parameters that are sampled that are the important parameters. We want to know that the important phenomena are being captured by the material that's been submitted by the applicant or the licensee. But second, we want to do exploratory analysis. We want to find out, are there any cliffs that we're going to fall off of. Are we going to do what if kind of tests. We want to know, is there some region that we can get into a lot of trouble. And this is pushing the state of the art with the codes often times.

And we also, from the operating reactor side, use the codes to resolve generic issues or to give us insights in the resolution of generic issues, such as the GSI-191 work that we reported and have been using the codes very heavily in. Within the Office of New Reactors, our primary working tool for accident analysis is the TRACE code. We've used the code extensively. We've asked the Office of Research to prepare an input model for each and every new reactor under review.

As part of that preparation, we've asked the Office of Research to prepare a code applicability report in which they will look at the individual design, are there unique features in that design and is the code capable of modeling those unique features in a reasonable manner? We want to then have, as a part of that report, a statement of is the plant model that they've provided reasonable? Does it give us a reasonable response? And these are all comparisons with whatever data are available.

We will then use the codes and look at the analysis for comparison and confirmation of the analysis that have been submitted by the applicant, but you have to keep in mind that when we do these confirmatory calculations, you cannot compare our calculation one-for-one with the calculation submitted by an applicant for a very important reason. The calculations we do in confirmatory space, are generally a founding calculation or a calculation with nominal conditions.

Remember I said just a minute ago that typically for a large break LOCA there are 30 to 35 parameters that are medium and high priority. These are the parameters that are sampled in doing a statistical analysis. We assign values for each of those parameters. The applicant, the vendor, samples those parameters, so that the calculation which they provide is their limiting calculation, their 95th percentile calculation as sample parameters where we have assigned values to those parameters. They are not one-to-one matches. So we cannot just directly compare the calculation which has been provided by the vendor or applicant with the calculations we get from the code, but the calculations should be reasonable enough to say, "Yeah, they've captured the right phenomena." We're predicting the same phenomena as the vendor is predicting and the timing is reasonably close.

MEMBER CORRADINI: So --

MR. LANDRY: Mike?

MEMBER CORRADINI: I didn't mean to stop you on this graph. I wanted to wait till you were done with this new graph to ask you a question, I'm sorry. So for code applicability reports, are there any completed code applicability reports for TRACE at this time?

MR. LANDRY: Yes. We have a code applicability report for ESBWR. We have --

MEMBER CORRADINI: Which we're discussing.

MR. LANDRY: Pardon me?

MEMBER CORRADINI: Which is the one we're discussing but I didn't know of any others.

MR. LANDRY: The reports have been prepared for EPR and they've been prepared for US APWR.

MEMBER BANERJEE: And what about AP-1000, was it all now finished?

MR. BAJOREK: We're -- this is Steve Bajorek. We've recently completed a draft version of the AP-1000 report.

MEMBER CORRADINI: And the other two that you mentioned, the US APWR and EPR, your subcommittee has?

MEMBER BANERJEE: No, we don't have it yet. I don't know whether it has come to us yet. I don't think so.

MR. BAJOREK: I don't think so. They are also fairly recent.

MR. LANDRY: We can -- we'll make sure that you get them.

VICE CHAIR ABDEL-KHALIK: Since your presentation is fairly high level, I will also keep my question at this stage at a high level. Are you aware of the January 11th, 2001 letter from ACRS to Chairman Meserve on the issues associated with industry developed from a hydraulic scope?

MR. LANDRY: Yes, but I haven't read it in a long time, so -- 2001 was a long time ago.

VICE CHAIR ABDEL-KHALIK: But we want to make sure that history doesn't repeat itself and I would assume that any feedback provided by this committee remains sort of active as along as it's relevant. So let me just focus on one of the issues raised in that letter, which says that many codes have the same ancestry including a 30-year old foundation. So given your description of how you're using TRACE, if you have a flawed tool that you're using to compare or assess the results of other flawed tools, what is it that you're really learning?

MR. LANDRY: The tool that we are using when we are using TRACE may have a lot of ancestry in common with TRAC, the TRAC family of codes, with the RELAP family of codes but there are major differences in the TRACE code. The numerics have been overhauled, significantly different numerics. It contains significantly different models, phenomenological models from some of the older codes. We would have to go model-by-model to talk about the differences and that, of course, is going to be proprietary because every vendor has taken the base codes and modified them is why part of the session today has to be proprietary for General Electric, because TRACG began as TRACB but has significantly different models than TRACB.

And the same is true for the RELAP5 code and for the WCOBRA TRAC code. All the codes have unique proprietary information contained. What we are using as a tool does not contain proprietary models. The TRACE code is open literature material. When we compare TRACE, we're comparing a code that has had a very extensive assessment program, hundreds of cases that have been used for assessing the code. We have a great deal of confidence in the code and we understand the code very well.

When we compare it with a vendor code, we are not, as I said, comparing one-to-one and when we look at our analysis versus their analysis, it is not an apples-to-apples analysis, comparison. But what we want to see is are the codes predicting their same phenomena at the same time or reasonably close? And repeatedly we see this, and it's not simply an artifact of we're all using 40-year old basic material, because our basic material has changed so dramatically to today that they may have started at the same point, but they've diverged significantly.

VICE CHAIR ABDEL-KHALIK: When you say they may have diverged, a concern has been raised over the past 35 years as far as I know over the momentum equation formulation in all of these codes and do you think or -- do you think that that shortcoming still remains?

MR. LANDRY: You're going to hear a great deal about that this afternoon.

VICE CHAIR ABDEL-KHALIK: Well, I just want to leap to that.

MR. LANDRY: All right. You're going to hear a lot about that from Joe and from Steve and yes, we've gone through this on every code in front of the committee. Graham Wallis has challenged us repeatedly on it. I'll let the research people address that because there's a lot more than saying are we comfortable with it or are we not comfortable with it? I think that they need to stand up here as they will as soon as I get down, and they'll start going into detail.

MR. BAJOREK: We'll go into that.

MR. LANDRY: So if you can hold that, that will be addressed.

MEMBER BANERJEE: I guess one point Said is making that it can be divided into two separate issues. One is because the codes share a common ancestry, they can model certain things and capture certain phenomena, but they may not be able to capture others because of that. And it doesn't mean that the phenomena is not important. It may exist in real life. For example, a code might allow you to have cold fluid on top of hot fluid. Now this is based in the structure of these codes because of the way they are.

So it doesn't mean that that's right. Obviously, cold fluid will not stay on top of hot fluid, but the codes will all predict that. So that's the danger of the common ancestry in some way. That within the structure of the model itself, there are shortcomings. And in a way, there's no way around this as long as you stay within this one dimensional framework. There are certain things you can capture and certain things you can't.

MR. BAJOREK: The codes have their limitations and their deficiencies. We tend to talk about the momentum equation issue. The one point, though, that I think needs to be made, even though some of the ancestry of these codes date back 30, 40 years, they have continually been improved by their assessments against much more recent data, STCF, CCTF, in the late '80s.

As we go to some of these advances plants, the APEX facilities, PUMA, these other facilities which have been designed, built and scaled specifically for the phenomena of these passive plants. That's why these codes have been assessed against that new data, to make sure they aren't going over a cliff or whether those numerics flawed as they might be, give you misleading answers.

MEMBER BANERJEE: Steve, it's not the numerics. It's the framework of the model itself.

MR. BAJOREK: It's the framework.

MEMBER BANERJEE: See, what you've got is a tapestry of conservation equations and empirical relationships. What you haven't changed are the set of conservation equations with their limitations in this type of formulation. You've change the empirical relationships there, largely and the solvers have improved a bit, but they're still way behind reality in other fields. So the real problem is that these empirical equations that you have or empirical relationships, surely have gotten better. I think that's really -- but the way you are using them is still somewhat limited by the framework that you've brought. And I think that's the point Said is trying to make, that you're limited in all the frameworks as far as similar and all the people have access to more or less the same experiments, so they've all sort of tuned their empirical relations to fit these experiments.

So if you're missing some important phenomena because of that framework, you know, how do you know? And if you look at the details of various experiments and so on, you are missing a lot of phenomena. We know that. Whether they are important or not is a separate issue. But, you know, so this really where we stand with that. But nonetheless, I think Ralph has given you a straight answer on this as to what they're doing without any -- why don't you carry on, Ralph and get to your --

MR. LANDRY: Okay, the last slide, in the Office of New Reactors, as I've said, we have been using TRACE as our primary work tool. We have input models for each of the new reactor designs. We're using the code extensively and today we have a high level of comfort with the code. We're very comfortable using it and we rely on it extensively. And with that, I just wanted to put into perspective the Office of New Reactors' view of the code and its use and then let the Office of Research now continue with talking details about the issues that have come up especially with respect to the ESBWR.

MEMBER CORRADINI: So from a user need standpoint, where does TRACE applicability end for NRO and another tool begins or do you look at TRACE as the complete tool you'll need for all portions of the advanced light water reactors? I mean, you're the user so I'm asking --

MEMBER BANERJEE: He's putting you on the spot.

MR. LANDRY: TRACE is the primary tool we use.

MEMBER CORRADINI: So that means for containment phenomena, TRACE is the primary tool?

MR. LANDRY: No, for containment of number of analysis that are being done with MELCOR.

MEMBER CORRADINI: But I'm kind of putting you on the spot, but purposely. I mean, if you said to me that TRACE is the necessary tool inside the reactor vessel and I come to some sort of boundary that turns into containment, and that's not where TRACE should do its best. X should do its best there, I understand that, but the way you said it at the end, I -- I heard a fuzzier answer than that.

MR. LANDRY: Well, we use other tools besides only TRACE.

MEMBER CORRADINI: Right.

MR. LANDRY: For example, one of the applicants, one of the vendors, is using a version of RELAP5-3D. The NRC does not have RELAP5-3D. That's a DOE code, but they have told us that their modified version is running in a RELAP5 mod 3 like manner. We don't understand what that is.

MEMBER CORRADINI: Neither do I.

MR. LANDRY: So we put together a model for their plant and we've run that model with RELAP5-3, RELAP5 mod 3, go to keep these threes and things straight here. We've run it with RELAP5 mod 3 for comparison with their RELAP5-3D like mod 3 mode. There was a feeling of is it truly operating in RELAP5 mod 3 like manner? But that was only for that comparison.

The other analyses that we're doing for comparative work, though for that plant are being done with the TRACE code. There was another design. They have a version of RELAP5 which we saw events occurring that we didn't believe. So we put together --

MEMBER BANERJEE: I think I know where you're going, Mike. Maybe --

MEMBER CORRADINI: Is Ralph --

MEMBER BANERJEE: Yeah, Ralph is the right person.

MEMBER CORRADINI: I think Ralph's the guy.

MEMBER BANERJEE: Okay, so I think he's leading into sort of --

MR. LANDRY: I'm giving you a awake-up of your topic today.

MEMBER BANERJEE: Yeah, GE is using TRACG for everything, okay, more or less. And they have one unified code that they are supporting for instability, for whatever, you know. And we're using this, that and the other and where do the boundaries change and how do we --

MR. LANDRY: We're doing that with TRACE also.

MEMBER CORRADINI: But I'll just ask you a pointed question and you don't have to answer it because I want to make sure that you see, because we can go off in many directions. What I guess I'm kind of asking is, you're the user. You have so many shekels to buy a resource. Are you going to spend your shekels on taking TRACE and making it work in containment or are you going to essentially just improve MELCOR so the boundary of the connection between TRACE and MELCOR is clear, defined and you can take yourself forward with audit calculations. That's kind of a nitty gritty way of asking the question.

MR. LANDRY: We make an effort to use the best tool available.

MEMBER CORRADINI: At any given time.

MR. LANDRY: At any given time.

MEMBER CORRADINI: Okay, fine.

MR. LANDRY: And we using the TRACE for the ESBWR for the LOCA. We're using it for ALOs. We're using it for ATWS. We're using it for stability.

VICE CHAIR ABDEL-KHALIK: Well, let me ask you a different kind of question. As a user you say you have high comfort level with TRACE. And I'm sure you are aware of the details of the assessments study that we will hear about later in which some areas were identified in which TRACE is judged to be less than adequate.

Are you, as a user of the code, confident that when you get results after using the code the you get adequate warnings telling you that you have used this code in a situation in which it was determined to be inadequate?

MR. LANDRY: Yeah, I think we -- we feel comfortable that the --

VICE CHAIR ABDEL-KHALIK: Do you get explicit warnings in the output telling you that you have run this code in a transient and part of the transient falls within the regime in which the code has been judged to be inadequate?

MR. LANDRY: If that occurs we get with our colleagues in Research and we discuss our the results reliable here or are they not? We don't just take our analysis at face value.

MEMBER BANERJEE: Are you getting warning signs like -- we know for example that in this report that ISL did, they say that there are regions where we recommend sensitivity studies be done because the code does poor prediction, what they call minimal.

MR. LANDRY: And we do that.

MEMBER BANERJEE: So now, what Said is asking for a general use. Are there sort of flags that tell you, you know, in this region we are now entering a situation where you've got problems and you should do sensitivity studies. Is that built into the quota? Do you have to read by, self-reported and make a list of these areas?

MR. LANDRY: A large part of that is the knowledge of the user. The analyst has to understand phenomenalogically, what is occurring and what the code is saying is occurring and not just treat it as a black box.

MEMBER BANERJEE: I guess there is nothing built into the code telling you that you're getting to some phenomena which is --

MR. LANDRY: I can't think of an example where that's occurring.

MEMBER BANERJEE: All right, I think that's the answer to your question.

MR. LANDRY: Maybe research can but --

MR. BAJOREK: There are some correlations where you're going out of bounds will be a warning but I think the short answer is, no, artificial intelligence has not been built into the use of these codes. At some point, you need to look at the transients, you need to look at results. You need to be aware of the code's shortcomings.

MEMBER BANERJEE: Well, because there are clear recommendations in this report from myself, where they tell you, you should do sensitivity analysis because the code is not reliable at minimum. You know, it says that straight out. And it should be in some user's guide somewhere, you know, because otherwise you have to read this report, which even I have a hard time reading. Imagine some poor user trying to read it.

VICE CHAIR ABDEL-KHALIK: Yeah, or some new staff member using the code. You're essentially assuming that this sort of institutional memory will be somehow automatically transferred to these young users and/or they'll go ahead and read that detailed assessment report that would allow them to find out whether or not you know, they have used the code in some part of the transient in which it has been judged to be inadequate.

MR. LANDRY: We don't give the code just to a new user. We train the people. The people are trained by research. We've -- we never have a new analyst sit and work with the code alone. They're always with somebody looking over their shoulder.

VICE CHAIR ABDEL-KHALIK: I assure you that if you get the most experienced analyst you have in NRO, and have him run the code, that they would not be aware of all -- each and every warning or constraint that is included in that assessment report.

MR. LANDRY: No, and we don't operate totally independently of the Office of Research either. We operate very closely with our colleagues in Research. We are a separate office, but we don't operate as though we're miles apart even though they are way up in the north country now.

MEMBER SHACK: They're allowed any code for anything that you'd trust to give to somebody and let them rip?

MEMBER POWERS: I'd give them MELCOR.

(Laughter)

VICE CHAIR ABDEL-KHALIK: Dana, you're biased. I think we are sort of slightly over time limit, so we should move on. So, thank you very much, Ralph, and I'm going to turn this back to Chris now. Is that Joe going to come up now? Okay.

MR. STAUDENMEIER: Okay, with this presentation, I'm going to try to give an overview in the open part of the meeting of the process we went through for TRACE applicability. I won't be able to get into details that will lead to proprietary conversations but hopefully it will give a flavor to everybody who can't stay for the closed part of the meeting and that's the process we go through and what we go through in determining that the code is applicable.

The first thing we do is look at design features of the plant, when we're looking at code applicability. For ESBWR it has the classic OBWR ECCS safety system strategy, isolate reactors on leak indications, activate ADS on low level signals or low inventory signals and then try to depressure the plant in a controlled manner to get to low pressure injection.

In the ESBWR there's no large liquid breaks in this plant like there are in operating jet pump plants which have the large recirculation plant down along the vessels there is no real large liquid break in this point to worry about. And as a result of that, there's no fuel cladding heat-up during any of the design basis LOCAs.

Unique safety system features of ESBWR has a gravity-driven cooling system instead of a pump cooling system and passive containment cooling system for long-term DKE removal. As part our ESBWR ECCS research program, we did a small amount of TRACE model development to add features we thought we needed in order to model the ESBWR. We performed a large amount of assessment. Some of the assessment was performed just for the base code that's applicable to all the plants and we developed a report to evaluate the assessment of the code and both integral and separate effects to determine that we were covering the conditions and the assessment was applicable to give us confidence that we're able to predict LOCAs in ESBWR.

And we also have a small confirmatory testing program that we called PUMA-E. It was a modification of our PUMA test facility which was originally a SBWR test facility. We did some modifications to make it more applicable to ESBWR. The document that kind of pulls all this together is a document that we're calling Adequacy of TRACE Version 5.0 for Simulating ESBWR Loss-of-Coolant Accidents. It was -- the work was done by a contractor pulling together all the assessments, reviewing documentation and putting this all into one report.

As a part of this, you have to determine what's important in calculating this plant. That's our process, that's been a tried and true process that we've applied when looking at applicability of codes to different power plants, review the TRACE code documentation to see what models are in the code and if they are applicable to the range of conditions that are going to appear in this plant.

Development of a consistent modeling approach for the plant analysis and test facilities to make sure you do assessments. They'll look good against the test data and make sure you're modeling the plant in an manner consistent with how you model the test facilities. We perform lots of TRACE code assessments against experimental data, analyze those and then as Ralph said, we also provide user guidelines and cautions for using TRACE for ESBWR applications.

MEMBER BANERJEE: Let me ask you a question. I mean, you sort of partially did a CSAU like methodology here but did I miss something related to the uncertainties in the report or is there a prompted in with uncertainty in prediction of your figure of merit?

MR. STAUDENMEIER: There's not a formal

uncertainty analysis for work in CSAU but we -- the parameters that we do think there are uncertainties in we'll do calculations to range them to see what the impact of it is, and I have one example of that and --

MEMBER BANERJEE: Well, you didn't put it all systematically together and say, you know, the predictions for your figure of merit are --

MR. STAUDENMEIER: No, we haven't done that.

MEMBER BANERJEE: Okay. And also if I recall, you've got a very large number of important phenomena in your approach and there was some talk from actually Tom Fletcher that you know, this should be probably narrowed down in some way. Of course, I don't recall how many but it was 400 or something?

MR. STAUDENMEIER: Yeah, I thought it was narrowed down to a hundred and some --

MEMBER BANERJEE: Yeah, even then it was rather large.

MR. STAUDENMEIER: Yeah, it's still large and what are really important, the number, I think is far less that that, but, yeah, it's something that would really need to be narrowed down.

MEMBER BANERJEE: So, would you say this report is like sort of a final document or does it still need some more work?

MR. STAUDENMEIER: Well, we do need to make some revisions to the document. If you recall in the meeting there were some PUMA calculations that needed to be updated to a later version that came about after the report was done and --

MEMBER BANERJEE: And that's been done. Are you going to show us that?

MR. STAUDENMEIER: That hasn't been finished yet, but --

MEMBER BANERJEE: Are you going to show us some comparisons with that?

MR. STAUDENMEIER: I have a comparison in the --

MEMBER BANERJEE: With PUMA-E, right?

MR. STAUDENMEIER: I don't have a PUMA-E comparison yet, but that will be done.

MEMBER BANERJEE: Because if recall, all your comparisons were with the PUMA experiments that were done at the end of the ESBWR program.

MR. STAUDENMEIER: That's correct.

MEMBER BANERJEE: And a lot of those had faulty instrumentation, off of range and things like that. So we have a whole lot of stuff with --

MR. STAUDENMEIER: There were some

instruments out of range. I don't --

MEMBER BANERJEE: Yeah, okay.

VICE CHAIR ABDEL-KHALIK: Does the assessment include the ability of the code to model any non-condensible gasses trapped within the lines?

MR. STAUDENMEIER: We don't have any assessments specifically dealing with trapping non-condensibles. There is a possibility of non-condensibles to be trapped in the experiments, so if it would happen, then in the experiment we would be comparing to the experimental results where they were trapped at that time.

VICE CHAIR ABDEL-KHALIK: But so far, this has not been done because there are no experiments. Is that what you're telling me?

MR. STAUDENMEIER: Well, we have experiments with ECCS lines that drain into the facility and have places where non-condensibles could be trapped. We didn't specifically try to trap non-condensibles in the line and see what happened when they drained out.

VICE CHAIR ABDEL-KHALIK: So let me ask the question again. Has this assessment included an assessment of whether or not the code can adequate model the performance of the gravity driven cooling system in the presence of non-condensible gasses in the lines?

MR. STAUDENMEIER: If non-condensible gasses go in the lines, they're going in the lines during the experiment and the code is modeling or trying to model what the experiment does. As I said, we didn't specifically try to trap them there. But if it happened in the test facilities, it happened and we try to calculate that as best we can.

MEMBER CORRADINI: Let me turn the question around, though. I think I know where Said is going but let me turn the question around. I'm not sure if you were here when we've had the ESBWR subcommittee meetings with the applicant but one of their outstanding items which I think they're somewhere in the process of delivering to staff is if the GDCS path which isn't supposed to, from the GDCS to the vessel, happens to have a plug of gas somewhere, how long will it take to clear or will it clear at all or how is the flow rate reduced from the flow. And we're waiting to hear how GE because they tried to track G calculation and we weren't satisfied. I guess that's the way it is.

What I'm asking is, if, as you as staff and an NRO will come in as a user need and say, "Okay, Research, we need help with the highly confident TRACE code that we want to use", can it do that calculation to somebody's satisfaction, eventually ours because if I'm not happy -- to be blunt, when we have the subcommittee meeting and I get the one from GE and I don't like it, I'm going to turn to staff and say, "What do you have that's better?" Is TRACE what's better?

MEMBER BANERJEE: I mean, this is a generic safety issue, right?

MEMBER CORRADINI: I mean, I picked the ESBWR but that's not the only one. We could go to the CMTs or the AP-1000. We can go to any sort of passive drainage system and this is kind of the thing that keeps on popping up and given low heads, you want to be clear that the drainage time and the flow rate is not severely adversely effected. And I guess I'm asking practically from a user need standpoint, can TRACE do this calculation?

MR. STAUDENMEIER: I think it could do the calculation. To really be confident, I think you'd need to specifically pull out or perform tests to show trapped gas purpose in the test and see --

MEMBER BANERJEE: Joe, there are experiments already in this area. It's clear that --

MR. STAUDENMEIER: No, it's really geometry specific. So you can't -- I don't think you can have generic experiments and --

MEMBER BANERJEE: Well, it's more than that. It's that at elbows, which is the real issue, you know, if you have a -- say a horizontal run and a vertical run or something, it's the elbow which becomes the limiting factor here. And you find that it floods. You know, you get full limitations earlier in that situation and if that happens, of course, your ability to deliver liquid goes down. And I guess that's the question that they're asking. If you have non-condensibles and they get trapped near elbows, do they actually give you problems with delivery of your GDTCS injection and can you model that? You'd probably have to put a specific flooding correlation in that. In that way you could probably do it. I think it can be done. It's probably not done in the code currently.

But this question came up in our subcommittee meeting, too. I think Said or somebody asked this specifically, because it had come up in the ESBWR meeting. So I think we know that you haven't done anything.

MR. BAJOREK: I mean most of the work that we've done with the non-condensibles is looking at its effect on the condensation process. There's a number of tests in there to take a look at its effect on condensation. There are some of the data from I think it's PANDA and some of the integral tests where we somewhat dance around the distribution in a large tank, okay, and how quickly it gets swept into that. But I think what you're really going after is this trapping of gasses in the horizontal safety injection lines.

MEMBER CORRADINI: Or just the -- but I'll give you a big picture. I'm back to Ralph, as the user, who has a need. Does he have high confidence in TRACE predicting the level in the vessel during the DBAs? And one of the reasons that this question comes up is, as I start losing inventory, I have to make up inventory. Is TRACE -- or is -- are we confident as to the rate at which the GDCS is making up inventory so that we've got a good prediction of where the water level is relative to the core?

That's it in a nutshell. I really don't care where the bubble goes, as long as there's enough water above the core. But that is a mechanism that could kind of clod up the whole situation.

MR. BAJOREK: I think in most of those

integral tests that's one of the critical parameters we tried to compare, how much flow are we getting from the DVI line, the DDCS.

MEMBER BANERJEE: But without a non-condensible bubble.

MR. BAJOREK: But without the non-condensible bubble.

MEMBER BANERJEE: Well, the problem really is this, basically these calculations are very simple. You've got some flow coming in and a hole there. The hole, you don't know what's going out, so you're sort of parametrically surveying it. So all you want to show is that when you parametrically survey different flow rates out of your hole, that you have enough flow coming in to keep the level up. Of course, if your flow goes down, eventually, the level will uncover the core. So it's as simple as that.

If your delivery goes down by a factor of two, you'll probably end up uncovering the core. So that's the issue. I mean, it's just a mass balance, the whole thing, which is a very simple calculation to do, which is why your scaling analysis boils the whole thing down to four or five equations if you look at it, and they predict everything almost correctly. I looked at it. And we did the same thing for AP-600.

It's pretty straightforward, really. Anyway, carry on.

MR. STAUDENMEIER: As one final comment on that, there are line losses specified in the GDSCS lines in the plant so that flow rate isn't very high going in. I mean, if you took those line losses that are specified out of the import, the vessel would fill up much faster than it would, so best estimate prediction of that without any orifice is in the line would show a lot faster filling of the vessel and if you look at the design basis accidents, there really is no way to get non-condensibles in the vessel and up the line before you get GDCS injection started.

VICE CHAIR ABDEL-KHALIK: So they can track a non-condensible gas between the squib valves, just depending on how we start up the plant.

MR. STAUDENMEIER: Oh, yeah, if it was in there ahead of time?

VICE CHAIR ABDEL-KHALIK: That's what I was saying, yeah.

MEMBER BANERJEE: That's really the issue.

VICE CHAIR ABDEL-KHALIK: So the question really is if you're cognizant of the fact that the applicant has been asked these questions during the ESBWR discussions, and if you are cognizant of the fact that the Applicant is trying very hard to evaluate this issue, why aren't you ahead of the curve?

MR. STAUDENMEIER: I guess one answer is we provided some data or pointed out some data where you are filling or draining a tank into an empty vessel where gas can get up into the line but that's -- we didn't do any specific testing to evaluate it.

MEMBER BANERJEE: Joe, we're running way behind, so let's move on.

MR. STAUDENMEIER: Okay.

MEMBER BANERJEE: Otherwise we'll be here till -- and my Chairman there will throw me out.

MR. STAUDENMEIER: This is just kind of a diagram of our adequacy determination process. Look at knowledge you have, let's get enough. Do you have applicable data?Knowledge of physics, what's important and go through this whole process to determine, see if the code is applicable to do these calculations.

Smaller development for ESBWR, the only model development we did was put a new film condensation model into the code, treats pure steam and mixtures of non-condensible gas in steam and it's applicable to PCCS, ICS tubes and containment walls. Assessment, first of all, we have assessment common to all reactors, separate effects void fraction, heat transfer, et cetera. That's document at our separate effects test manual. It's available in ADAMS. All our manuals are available in ADAMS.

Specific assessment for a film condensation component test, GE full scale PCCS and ICS tests. And integral tests applicable to ESBWR. Our PUMA-E integral testing for ESBWR it's designed as study system interactions of the ESBWR safety systems. This is kind of a drawing of what PUMA looks like. It's actually a lot more -- if you've actually been there, it's not as open as it looks in this picture. There's other things and structures holding it all up that make it a little hard to get around.

And as a result of the PUMA-E testing, we think the ESBWR and SBWR behavior is qualitatively the same PUMA-E tests and PUMA SBWR test behaved in the same sort of way we think we understand the behavior of the safety systems.

MEMBER CORRADINI: So even though we're not going to see the details of that, the last thing you said, your evaluation of the calculations relative to PUMA-E will make it seem reasonable? I mean, I thought Sanjo's questions earlier we were going to see the PUMA-E results and I thought your answer was no, and I --

MR. STAUDENMEIER: I have a couple of PUMA-E results to show.

MEMBER CORRADINI: Okay, excuse me.

MR. STAUDENMEIER: We don't have TRACE assessment. ESBWR ECCS calculations, we performed calculations for a range of break sizes and locations from the top of the vessel, the steam line break, which is a fairly large steam break, down to the lowest elevation bottom drain line break, which is a liquid break. Performed sensitivity calculations to examine the effect of model deficiencies and uncertainties and concluded a calculated response of the ESBWR ECCS is predicable and consistent with integral test results.

And by predictable I mean, things are understandable. They go in one direction. There's no funny cliffs or things like that, that we can fall off of or no oscillations or out of expected behavior. That should be predictable and it is acting consistently with our tests as expected.

Conclusions, calculated performance and response, ESBWR ECCS is predictable and consistent with integral test results and TRACE is adequate as an audit tool for analyzing the ESBWR ECCS system response. Now, one thing I'd also like to add, in cases where we don't think TRACE is applicable, there are studies that are done with CFD to supplement our system codes or experimental -- we'd want to take experimental data, so we don't just say TRACE is good for everything. We look for situations that we don't think it's giving good predictions or maybe cases where we need to back up the predictions by something more detailed and in that case we have done CFD analysis where we think it's applicable.

MEMBER CORRADINI: What are two or three example of that?

MR. STAUDENMEIER: Well, for ESBW -- well, for ESBWR there's something looking at mixing in the downcomer under shutdown conditions and looking at by-pass of cooling flow coming in and then going back out just because of the way the pipes are located to see if you're bypassing too much of the RHR cooling flow to make sure you're keeping the reactor cool during shutdown or CFD calculations done to evaluate that situation.

MEMBER CORRADINI: So those are the two key ones?

MR. STAUDENMEIER: That's one for ESBWR that I am aware of for -- I mean, generally it's situations of mixing or flow in open regions where CFD is done that it's largely single-phase flow. We haven't done two-phase. There was one case in a chimney I think where a two-phase calculation was done.

MEMBER CORRADINI: Well, I mean, I'm not going to disagree with you because I'm trying through cruise back to the report to catch up on certain other things, but I guess your two examples are reactor vessel like which makes me feel good, because I expect that's where TRACE -- but yet in the ISL assessment, their examples of where they're concerned are all containment modeling issues. So there's nothing in containment that you feel that TRACE needs to be backed up by something else?

MR. STAUDENMEIER: Well, in the next presentation I'll show it's backed up by what we think is a bounding calculation for containment back pressure.

MEMBER CORRADINI: Okay, fine.

MEMBER BANERJEE: Now, TRACE is supposed to be operated like a best estimate code, right?

MR. STAUDENMEIER: That's one mode of operation, probably the most vitally used mode of operation.

MEMBER BANERJEE: Yeah, in which case you do need to have an evaluation of uncertainties but do you model uncertainties due to input uncertainties?

MR. STAUDENMEIER: And I mean, there are input bounding assumptions that can be made like flow rate at a pressure specified for a relief valve or something like that.

MEMBER BANERJEE: Right, but you have model uncertainties like your condensation rate on the GDCS pool surface is too high. Clearly, the non-condensibles that build up there and prevent you know, which the steam has to diffuse through the non-condensibles being heavier, you know, give you uncertainties as to what the temperature of the DGCS pool should be.

You know, you predict a higher temperature or slightly than is there. So there re a lot of model uncertainties as well and I haven't seen a systematic evaluation. Even though that report does do quite a bit of sensitivity analysis, nowhere is it all put together and, you know, say, "This is what we expect, this is amount of uncertainty. This is due to input, this is due to the models". That would give me a much more comfortable feeling about the code. I mean, it can't be all things but at least we should know what the uncertainties are.

And that's sort of missing. And it's there in some implicit ways if you read carefully but it's not all put together somewhere.

MEMBER APOSTOLAKIS: Are there any plans to do this?

MEMBER BANERJEE: That's what I was asking.

MR. BAJOREK: Yes, right now, TRACE, I'll refer to it as a -- we try to treat it as a realistic code, but then full best estimate treatment would mean going through, looking at the PIRT, ranging all of the uncertainty parameters. We have started that work. We're looking at doing that for conventional plants and also making the methodology as such that we could extend it to the passive plants, but we are not there yet. We've just gotten started with that.

MEMBER BANERJEE: Okay, I think we can move on to closing the session then. We're running behind time, so this is -- I guess is anybody --

MEMBER CORRADINI: Anybody that is now here that should not be here, please leave; is that what you're saying.

MEMBER BANERJEE: Yeah. Those who are not GE or NRC.

(Recessed to move to closed session.)

BEAVER VALLEY POWER STATION
License Renewal Application

FENOC
Presentation to ACRS

July 8, 2009

FENOC

FirstEnergy Nuclear Operating Company

Introductions

- **Pete Sena, Site Vice-President**
- **Mark Manoleras, Site Engineering Director**
- **Cliff Custer, License Renewal Project Manager**
- **John Thomas, Project Technical Lead**
- **Site Subject Matter Experts and members of the LRA core team**

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FirstEnergy Nuclear Operating Company

Agenda

- **Site Description**
- **License Renewal Project**
- **Open Item Resolution - Inaccessible Medium-Voltage Cables**
- **Subcommittee Follow-Up Items**
  - Recent OE: MRP-146 Inspections
  - Containment Liner
- **Summary**

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Site Description

- **Two unit , 3-Loop, nominal 2900 MW_{th} Westinghouse PWR**
- **17 miles west of McCandless, PA, on the Ohio River**
- **Owned/operated - Ohio Edison and Toledo Edison / FirstEnergy Nuclear Generation Group**
- **BV-1 Commercial in 1976; BV-2 in 1987**

FENOC

FirstEnergy Nuclear Operating Company

License Renewal Project

- **BVPS core team remained engaged with industry and NRC**
- **Independent assessments by industry panel, site QA, peer review group, and FENOC Corporate Nuclear Review Board**
- **Methodology consistent with NEI 95-10**
- **Project intent to maximize GALL consistency**
  - 91.8% of AMR line items used notes A-E (GALL consistent)

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FirstEnergy Nuclear Operating Company

Open Item Resolution

Inaccessible Medium-Voltage Cables

- **Open Item 3.0.3.1.11-1 is Closed**
- **AMP consistent with GALL**
- **New commitment:**
  - Methodology to demonstrate cables will continue to perform their intended function,

    -or-
  - Minimize exposure to significant moisture,

    -or-
  - Replacement of cables

FENOC

FirstEnergy Nuclear Operating Company

Subcommittee Follow-Up Item

Recent OE: MRP-146 Inspections

- **Inspections per FENOC LR Commitment No. 31**
- **13 BV-1 piping locations “screened-in” and examined during 1R19 (Spring 2009)**
- **NDE indications on 2" diameter RCS "A" loop drain line**
- **Evaluation probable cause “Thermal Fatigue”**
  - Metallurgical confirmation pending
- **Replaced the pipe containing the indication**

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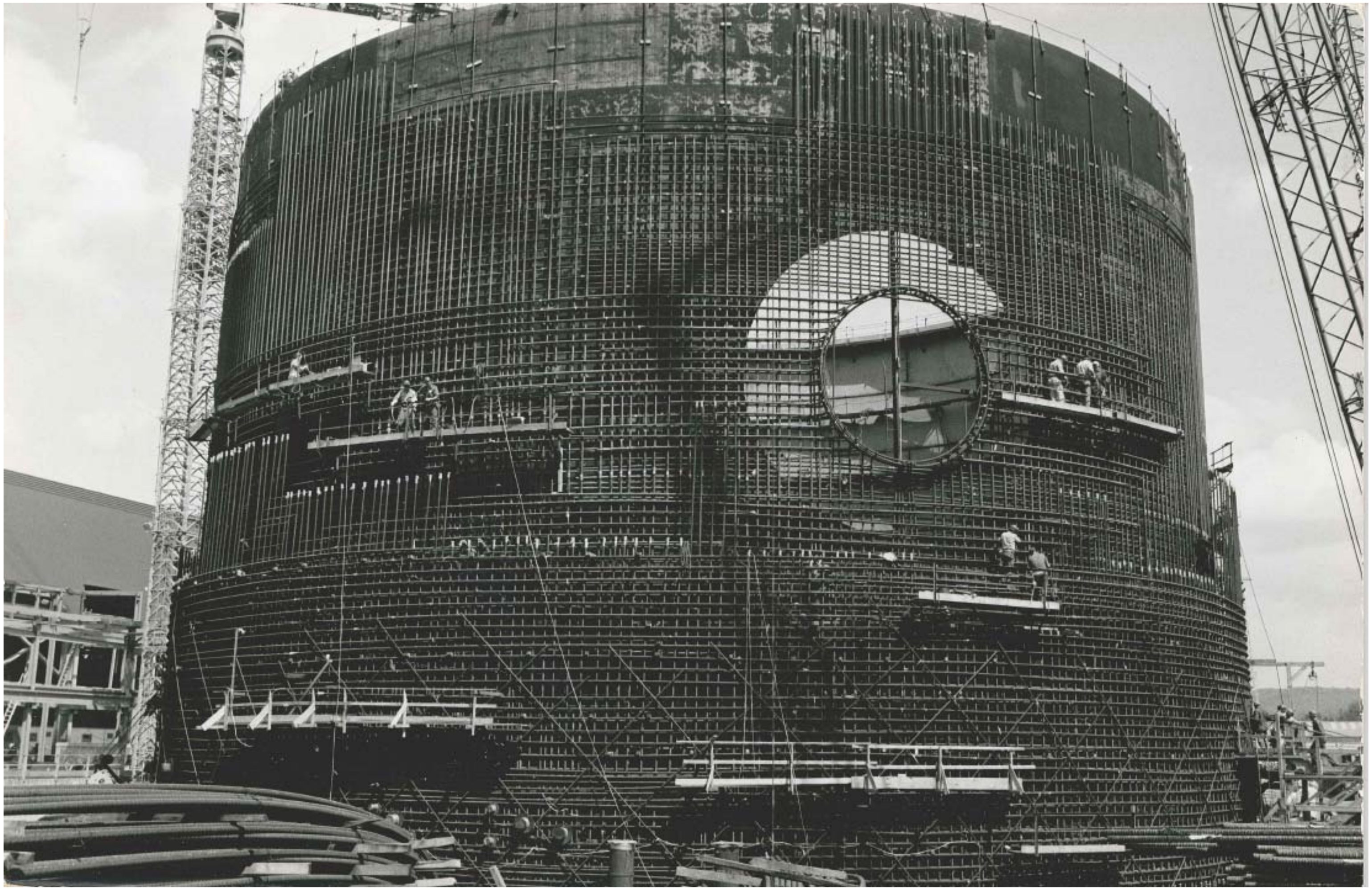
Subcommittee Follow-Up Item



BV-1 Containment Construction

FENOC

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Subcommittee Follow-Up Item

Containment Liner Design

- **Carbon steel liner; nominal thickness: 1/4 in. floor, 3/8 in. wall (cylinder), 1/2 in. dome.**
- **"Insert" plates (5/8 in. to 1-1/2 in. thick) have separate studs to transfer large component loads.**
- **"Overlay" plates (3/8 in. to 5/8 in. thick) are welded to liner over sufficient liner studs to transfer lighter loads.**
- **Penetrations transfer loads directly to concrete in wall.**

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Subcommittee Follow-Up Item

Containment Liner - 2006

- **During 1R17 concrete removal by hydro-demolition exposed back side of liner during creation of opening for S/G replacement**
- **Three areas of corrosion were found on the concrete side of the exposed liner**

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Subcommittee Follow-Up Item

Containment Liner OE – 2009

- **IWE visual inspection identified paint blister with rusting**
- **Cleaning revealed primer coat blistering and a small through-wall flaw (1 in. x 3/8 in.)**
- **UT exams determined extent of corrosion around the flaw (approx. 2 in. x 5 in.)**

Subcommittee Follow-Up Item

Containment Liner OE – 2009 (Cont)

- **Wood found embedded in concrete immediately behind liner**
- **Lab analysis confirmed moisture in wood**
- **Corrosion attributed to foreign material (wood) in contact with liner in presence of moisture**
- **Concrete found in good condition**
- **Replaced affected section of liner**

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Subcommittee Follow-Up Item

Containment Liner - 2009

- **Corrective Actions**
  - Follow-up UT of replaced area in next Unit 1 refueling outage.
  - IWE Visual examinations
    - Next BV-1&2 Refueling Outages
    - Normally scheduled IWE exams for the following outage
  - Supplemental volumetric examinations will be performed on both Unit's containment liners prior to the period of extended operation. If degradation is identified, the degraded area(s) will be evaluated and follow-up examinations will be performed to ensure the continued reliability of the containment liner.

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Closing Remarks

Aging management programs and related commitments provide reasonable assurance that aging will be managed such that SSCs will continue to perform their intended functions during the period of extended operation.

FirstEnergy Nuclear Operating Company

Beaver Valley License Renewal

FENOC

FirstEnergy Nuclear Operating Company

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee

Beaver Valley Power Station, Units 1 and 2

Safety Evaluation Report

July 8, 2009

Kent Howard, Project Manager
Office of Nuclear Reactor Regulation

Overview

- License Renewal Application (LRA) submitted August 27, 2007
- Westinghouse 3-Loop – PWR
- 2900 megawatt-thermal, each unit
- Operating license DPR-66 (Unit 1) expires January 29, 2016
- Operating license NPF-73 (Unit 2) expires May 27, 2027
- Located approximately 17 miles west of McCandless, PA

Overview

Recap of February 2009 ACRS sub-committee meeting

- SER with open item issued January 9, 2009
- One (1) Open Item
- No Confirmatory Items
- 249 RAIs Issued
- 31 Commitments (Unit 1)
- 32 Commitments (Unit 2)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Overview

Summary of February 2009 ACRS sub-committee meeting follow-up items

- **Inaccessible Medium Voltage Cable**
  - Open Item 3.0.3.1.11-1
  - Suitability of cables for submergence
- **Containment Liner**
  - Function of the containment liner
  - Evidence of corrosion
- **Boral**
  - New program, submitted after issuance of SER w/ Open Item
- **Metal Fatigue/Cycle Count Histograms**
  - Questions on methodology used to count Unit 1 transients

Overview

Subsequent to sub-committee meeting

- 6 additional RAIs were issued
- Resolved Open Item 3.0.3.1.11-1 related to Inaccessible Medium-Voltage Cables
- Additional commitment added for Unit 1 and Unit 2
- Recent containment liner issue was addressed in Final SER which was issued on June 8, 2009

Section 3: Aging Management Review Results

Open Item 3.0.3.1.11-1

- Staff was concerned that inaccessible medium-voltage cables that have been submerged for a period of time may be degraded and may not perform their intended function during the period of extended operation.

Section 3: Aging Management Review Results

- Resolution
  - At Subcommittee meeting, the applicant had plant-specific AMP, Inaccessible Medium-Voltage cables.
  - Program was revised to be consistent with GALL XI.E3

Section 3: Aging Management Review Results

Commitment #11 (Unit 1)/Commitment #12 (Unit 2)

(1) Adopt an acceptable methodology that demonstrates cables will continue to perform their intended function, OR;

(2) Implement measures to minimize cable exposure to significant moisture through dewatering manholes, OR;

(3) Replace the in-scope, continuously submerged medium-voltage cables with cables designed for submerged service.

Section 3: Aging Management Review Results

- Boral
  – Boral Surveillance Program AMP (Unit 1) (B.2.43) added to LRA.

    - The new program was evaluated by NRC staff and determined that the applicant demonstrated that the effects of aging will be adequately managed as required by 10 CFR 54.21(a)(3)

Section 3: Aging Management Review Results

- Containment Liner Issue
  - On April 23, 2009, during a scheduled Unit 1 IWE inspection, a paint blister was discovered on the containment liner, revealing through-wall corrosion.
  - Staff issued RAI B.2.3-4 on May 7, 2009 requesting the applicant explain how the recent plant specific operating experience would be incorporated into the IWE AMP.

Section 3: Aging Management Review Results

Actions Taken to Address Issue

Unit	Date	Activity
2	Next Outage	100% visual exam of liner plate
1	Next Outage	100% visual exam of liner plate
1	Next Outage	UT repaired area
2	Subsequent Outage	Scheduled IWE visual examination of liner plate
1	Subsequent Outage	Scheduled IWE visual examination of liner plate
1	Prior to start of PEO	Volumetric exam of 75 1’ x 1’ areas of liner plate to ensure 95% confidence level
2	Prior to start of PEO	Volumetric exam of 75 1’ x 1’ areas of liner plate to ensure 95% confidence level

License Renewal Inspections

Dr. Ronald Bellamy

Region I DRP Branch Chief

Medium Voltage Submerged Cables and Containment Liner

Medium Voltage Submerged Cables

- Regional Inspection in June 2009
- Inspection identified safety related cables were not qualified for *continuous* submerged conditions
- FENOC took immediate & long term corrective actions
- Results will be documented in next Resident inspection report

Containment Liner

- Regional Specialist on site during spring 2009 Unit 1 outage. Inspection Report 2009-006 discusses acceptability of liner repair

Section 4: Time-Limited Aging Analyses

4.3 Metal Fatigue

- 2003 Cycle counts
  - Conservative results from a previous Westinghouse analysis
- 2009 Reconstitution of cycles
  - Addressed heatups (cooldowns) for each unit
  - Covered entire plant operating history
  - Utilized monthly operating reports and control room data
- NRC staff review (audit)
  - Covered applicant's analysis of data from 1996 and 1997, when the new cycle counts were reduced
  - No issues identified with applicant's approach

Conclusion

- On the basis of its review, the staff determines that the requirements of 10 CFR 54.29(a) have been met.

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Questions

License Conditions

- The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the UFSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.
- The second license condition requires future activities identified in the UFSAR supplement to be completed prior to the period of extended operation with the exceptions as follows: For BVPS-1: UFSAR Supplement Commitments 20, 24, 29, and 31. For BVPS-2: UFSAR Supplement Commitments 22, 28, and 32.
- The third license condition requires that all capsules in the reactor vessel that are removed and tested meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

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Division 1

DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1175

(Proposed Revision 3 of Regulatory Guide 1.100, dated June 1988)

SEISMIC QUALIFICATION OF ELECTRIC AND ACTIVE MECHANICAL EQUIPMENT AND FUNCTIONAL QUALIFICATION OF ACTIVE MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

This guide describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment for nuclear power plants (NPPs).

For seismic qualification of electric and active mechanical equipment, the general requirements appear in Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the *Code of Federal Regulations* (10 CFR Part 50) (Ref. 1) and 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." (Ref. 2) Particular sections include General Design Criterion (GDC) 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Dynamic Effects Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50; Criterion III, "Design Control," Criterion XI, "Test Control," and Criterion XVII, "Quality Assurance Records," of

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This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position.

Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemaking, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; emailed to NRCREP@nrc.gov; submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; faxed to (301) 415-5144; or hand-delivered to Rulemaking, Directives, and Editing Branch, Office of Administration, US NRC, 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by July 11, 2008.

Electronic copies of this draft regulatory guide are available through the NRC's interactive rulemaking Web page (see above); the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML072620346.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; and Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50.

Section III, "Definitions," of Appendix S to 10 CFR Part 50 states that the structures, systems, and components (SSCs) required to withstand the effects of the safe-shutdown earthquake (SSE) ground motion or surface deformation are those necessary to assure (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1). Section IV(a)(1)(ii) of Appendix S to 10 CFR Part 50 requires that the NPP must be designed so that, if the SSE ground motion occurs, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, the design of these safety-related SSCs must take into account applicable concurrent normal operating, functional, and accident-induced loads. Section IV (a)(1)(iii) of Appendix S to 10 CFR Part 50 requires that the safety functions of SSCs must be assured during and after the vibratory ground motion associated with the SSE ground motion through design, testing, or qualification methods.[1]

The general requirements for functional qualification of active mechanical equipment also appear in 10 CFR Part 50 and 10 CFR Part 52. In 10 CFR Part 50, particular sections include GDC 1, "Quality Standards and Records," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 30, "Quality of Reactor Pressure Boundary," GDC 37, "Testing of Emergency Core Cooling System," GDC 40, "Testing of Containment Heat Removal System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," GDC 46, "Testing of Cooling Water Systems," and GDC 54, "Piping Systems Penetrating Containment," of Appendix A to 10 CFR Part 50, as well as Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50.

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the NRC regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required.

This regulatory guide contains information collection requirements covered by 10 CFR Part 50 and 10 CFR Part 52, and that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays currently valid OMB control numbers.

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[1] Appendix S to 10 CFR Part 50 applies to applicants for a design certification or combined license pursuant to 10 CFR Part 52 or a construction permit or operating license pursuant to 10 CFR Part 50 after January 10, 1997. However, the earthquake engineering criteria in Section VI, "Application to Engineering Design," of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," continue to apply for either an operating license applicant or holder with a construction permit issued before January 10, 1997.

B. DISCUSSION

Background

The NRC issued Revision 2 of Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” (Ref. 3) in June 1988. With a few exceptions and clarifications, it endorsed the Institute of Electrical and Electronics Engineers (IEEE) Std 344-1987, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,” (Ref. 4) and extended the application of that standard to the seismic qualification of mechanical equipment. In extending the application of IEEE Std 344-1987 to mechanical equipment, the NRC staff recognized differences in seismic qualification methods for electric equipment[2] (including instrumentation and control (I&C) components) and mechanical equipment. Specifically, Revision 2 of this regulatory guide stated that seismic qualification of mechanical equipment by analysis is permitted when such equipment can be modeled to adequately predict its response. Revision 2 also stated that the American Society of Mechanical Engineers (ASME) was developing a standard for seismic qualification of mechanical equipment and, upon publication of that standard the NRC staff would review it for suitability for endorsement by a revision of this regulatory guide.

In 1981, the NRC issued Revision 0 of Regulatory Guide 1.148, “Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants.”~~.~~ (Ref. 5)~~.~~ With a few exceptions and clarifications this guide endorsed American National Standards Institute (ANSI) N278.1-1975, “Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard.”~~.~~

In 1994, the ASME issued a standard, ASME QME-1-1994, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” (Ref. ~~5~~6) This ASME standard eventually replaced the ANSI N278.1 standard. The ASME QME-1 standard covers both seismic qualification and functional qualification of active mechanical equipment. The ASME subsequently revised and reissued the standard in 1997, 2000, and 2002, with the last revision issued in November 2007 as ASME QME-1-2007 (Ref. ~~6~~7). Furthermore, the IEEE updated IEEE Std 344-1987 and issued it as IEEE Std 344-2004 (Ref. ~~7~~8) in June 2005.

The NRC developed this regulatory guide (i.e., Revision 3) to endorse, with exceptions and clarifications, the IEEE Std 344-2004 and the ASME QME-1-2007. (This is the first time the NRC is endorsing ASME QME-1.)~~.~~ This revision of the regulatory guide will also subsume Regulatory Guide 1.148. Regulatory Guide 1.148 is intended to be deleted when Revision 3 of Regulatory Guide 1.100 is approved. ~~(This is the first time the NRC is endorsing ASME QME-1).~~ Specifically, Sections B.-1 and C.-1 of this regulatory guide endorse, with exceptions and clarifications, the entire IEEE Std 344-2004 and Section QR, "General Requirements," and Nonmandatory Appendix QR-A, "Seismic Qualification of Active Mechanical Equipment," of ASME QME-1-2007 for the seismic qualification of electrical and active mechanical equipment. ~~Specifically, Sections B.1 and C.1 of this regulatory guide endorse, with exceptions and clarifications, the entire IEEE Std 344-2004 and Section QR, “General Requirements,” and Nonmandatory Appendix QR-A, “Seismic Qualification of Active Mechanical Equipment,” of ASME QME-1-2007 for the seismic qualification of electric and active mechanical equipment.~~

[2] Hereafter in this regulatory guide, the term “electric equipment” means an assembly of electric and electronic components designed and manufactured to perform specific functions, and the term “electric component” or “electronic component” means items from which the equipment is assembled (e.g., resistors, capacitors, wires, connectors, microprocessors, switches, springs, and instrumentation and control items).

~~respectively.~~ Sections B.2 and C.2 of this regulatory guide endorse, with exceptions and clarifications, Section QR and the remaining sections of ASME QME-1-2007 (except Non-mandatory Appendix QR-A) for the functional qualification of active mechanical equipment. The ASME QME-1 standard defines active mechanical equipment as "Mechanical equipment containing moving parts, which, in order to accomplish its required function as defined in the Qualification Specification, must undergo or prevent mechanical movement. This includes any internal components or appurtenances whose failure degrades the required function of the equipment."

1. Seismic Qualification of Electric and Active Mechanical Equipment

The major change from IEEE Std 344-1987 to IEEE Std 344-2004 is the update and expansion of Clause 10, "Experience," which describes the use of experience data as a method for seismic qualification of Class 1E electric equipment (including I&C components). Experience data include earthquake experience data and test experience data. Nonmandatory Appendix QR-A to ASME QME-1-2007, which has been updated and expanded from Nonmandatory Appendix QR-A to ASME QME-1-2002, also includes the use of experience data as a method for seismic qualification of active mechanical equipment.

The use of earthquake experience data for seismic qualification of electric and mechanical equipment has its origin in the NRC research program associated with Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Mechanical and Electric Equipment in Operating Nuclear Power Plants." In 1980, the NRC staff raised a safety concern that licensees had not conducted the seismic qualification of electric and mechanical equipment in some older vintage NPPs (i.e., plants with construction permit applications docketed before about 1972) in accordance with the licensing criteria for seismic qualification of equipment acceptable at that time (i.e., IEEE Std 344-1975 (Ref. ~~8~~9) and Regulatory Guide 1.100, Revision 1, (Ref. ~~9~~10) issued August 1977). Therefore, equipment in the older vintage NPPs may not have been adequately qualified to ensure its structural integrity and/or proper functionality in the event of an SSE ground motion. As a result, the NRC established the USI A-46 program in December 1980 and, in February 1987, issued Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electric Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," (Ref. ~~10~~11) to address this safety concern. The NRC staff categorized approximately 70 NPP units in the United States as "USI A-46 plants."

In 1982, the Seismic Qualification Utility Group (SQUG) undertook the development of an earthquake experience and a test experience database to address USI A-46. Because of the scarcity of data on equipment that was subjected to strong earthquake motion in US NPPs~~Because of the scarcity and low intensity level of earthquakes to which U.S. NPPs were exposed~~, the SQUG and its contractors performed a pilot study to determine the feasibility of using actual earthquake experience data from nonnuclear plants located worldwide (e.g., fossil-fueled power plants, substations, petrochemical plants) and existing test experience data from domestic NPPs to evaluate the performance of electric and mechanical equipment in those facilities to infer the susceptibility of similar NPP equipment to seismic loads. The SQUG concluded, and the NRC agreed, that the use of experience data was feasible for the purpose of verifying equipment seismic adequacy for the older vintage USI A-46 plants. The -staff does not accept the use of SQUG guidelines for seismic qualification of equipment in non USI A-46 plants licensed under 10CFR–Part 50 or in plants licensed under 10CFR -Part 52.

Large uncertainties exist in the seismic qualification of equipment as a class, on the basis of earthquake experience data because (1) it is difficult to compile a credible earthquake experience database (e.g., estimation of ground and floor earthquake excitations used in the earthquake experience database); (2) the inclusion rules and exclusion rules (termed "prohibited features" in IEEE Std 344-2004) of equipment in the database may not be complete; (3) the similarity between equipment in fossil/petrochemical plants in the database and those in NPPs is difficult to establish; and, most importantly, (4) generally there is not sufficient credible information from the earthquake experience database to provide assurance of the proper functionality of certain active electric equipment during earthquakes. ~~Active electric equipment is the equipment that must either undergo a change of state or prevent a change of state in order to accomplish its required functions. The NRC staff is particularly concerned about the use of earthquake experience data for seismic qualification of active electric equipment that~~

In using the test experience data for seismic qualification of electric equipment, quantifying the damage potential of equipment under testing should capture the combination of input motion and the equipment item exhibiting a particular malfunction. Consideration should be given to the likelihood that the resonant frequency for items of equipment of the same class ~~may inadvertently change state during an earthquake such that its intended safety functions are not performed during and/or after an earthquake. Examples of such active electric components are certain types of relays, contactors, circuit breakers, switches, sensors, and potentiometers.~~

differ significantly, thus, multiple malfunction mechanisms for components and subcomponents need to be considered in comparing the test response spectra (TRS) and the required response spectra (RRS).~~The NRC has three concerns regarding the use of test experience data for the seismic qualification of electric equipment. First, it is difficultcare must be given to quantifying the damage potential of equipment under testing since it depends on the combination of input motion and the equipment item exhibiting a particular malfunction. Furthermore, Tthe resonant frequency corresponding to a given malfunction is mostly unknown, and this frequency for each item of equipment of the same class can differ significantly. There could be multiple malfunction mechanisms for components and subcomponents that need to be considered in comparing the test response spectra (TRS) and the required response spectra (RRS). This can lead to uncertainties. These uncertainties should be accounted for. Therefore, an equipment capacity factor has to be considered to cover the uncertainties in high-level testing for an equipment class. 12. Second, the~~

The technologies and designs of certain electric components (such as certain types of relays and microprocessor-based components) have undergone significant changes since the NRC issued Revision 2 of this regulatory guide, as a result of the more prevalent use of digital I&C components in place of the traditional analog I&C components. Some solid-state relays and microprocessor-based components ~~are quite fragile in terms of withstanding~~may be sensitive to earthquake excitations. The staff considers the use of test experience data from the older-vintage electric components of this type for the seismic qualification of the new generation of such electric components to be inappropriate and unacceptable. ~~Third~~Furthermore, since no new NPPs were built in the USA after the early 1980s, a number of manufacturers for electric or active mechanical equipment are no longer in business, and the appropriateness of using the test experience of old equipment made by manufacturers no longer in business for the seismic qualification of modern equipment designs made by different manufacturers is highly questionable.

~~The NRC staff has two other concerns as well. The NRC staff has a concern regarding electric and active mechanical equipment exposed to harsh environments, aging, and earthquakes. In such cases, the NRC staff does not find it acceptable to use experience data (earthquake or test experience data) for seismic qualification of equipment. The test sample shall be subjected to simulated operating-basis earthquake (OBE) and SSE seismic vibrations in accordance with IEEE Std 344-2004.~~

~~Another NRC staff concern is the high-frequency earthquake ground motion for certain plant sites.~~ Recent studies related to the early site permit applications at certain hard-rock-based plants along the east coast of the United States indicated that the site-specific spectra may exceed the certified design spectra of those new plants in the high-frequency range (20 hertz (Hz) and above). This exceedance cannot always be eliminated, even with incoherency added to the soil-structure interaction analyses. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. For operating boiling-water reactor (BWR) plants, the seismic qualifications of some safety-related electric and active mechanical equipment were performed using IEEE-344-type tests with intentional high-frequency contents to account for concurrent BWR hydrodynamic loads. However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz, although the TRS may have shown a zero period acceleration (non amplified frequency range) up to 100 Hz~~although the TRS may have shown a zero period acceleration (ZPA) up to 100 Hz~~. Ball joints and kinematics linkages of the shake tables generated these inadvertent high frequencies~~, which the NRC staff considers to be noise signals~~ that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies. Therefore, any attempt to use such past test experience data for the seismic qualification of high-frequency-sensitive equipment or components in such a plant is not appropriate unless the frequency content of the power spectral density (PSD) of the test waveform has been evaluated in accordance with Annex B of IEEE 344-2004.~~Therefore, any attempt to use such past test experience data for the seismic qualification of high-frequency-sensitive equipment or fragile components in such plants clearly is not appropriate.~~ When new seismic qualification tests are planned for equipment in such plants, the formulation of the test input waveforms should properly consider this high-frequency ~~ground motionmotion~~excitation~~ concern~~.

2. Functional Qualification of Active Mechanical Equipment

The ASME QME-1-2007 standard describes requirements and guidelines for qualifying active mechanical equipment used in NPPs. The foreword to the standard indicates that the standard may be applied to future NPPs or existing operating NPP component replacements, modifications, or additions, as determined by regulators and the NPP licensees. The ASME QME-1-2007 provides functional qualification guidance for nonmetallic parts, dynamic restraints, pumps, and valves. The following sections and appendices of ASME QME-1-2007 provide the functional qualification guidance for this active mechanical equipment—(1) Section QR, (2) Nonmandatory Appendix QR-B, “Guide for Qualification of Nonmetallic Parts,” (3) Section QDR, “Qualification of Dynamic Restraints,” and its Nonmandatory Appendices QDR-A, “Functional Specification for Dynamic Restraints,” QDR-B, “Restraint Similarity,” and QDR-C, “Typical Values of Restraint Functional Parameters,” (4) Section QP, “Qualification of Active Pump Assemblies,” and its Nonmandatory Appendices QP-A, “Pump Specification Checklist,” QP-B, “Pump Shaft-Seal System Specification Checklist,” QP-C, “Pump Turbine Driver Specification Checklist,” QP-D, “Pump Similarity Checklist,” and QP-E, “Guidelines for Shaft-Seal System Material and Design Consideration,”~~ and~~ (5) Section QV, “Functional Qualification Requirements for Active Valve Assemblies for Nuclear Power Plants,” ~~and~~ its Mandatory Appendix QV-~~1~~II, “Qualification Specification for Active Valves,” and its Nonmandatory Appendix QV-A, “Functional Specification for Active Valves for Nuclear Power Plants,” and (6) Section QV-G, “Guide to Section QV: Determination of Valve Assembly Performance Characteristics.” ~~“Qualification Specification for Active Valves.”~~ The major change from ASME QME-1-2002 to ASME QME-1-2007 in terms of the functional qualification of mechanical equipment is a complete rewrite of Section QV and the new Mandatory Appendix QV-I~~1~~.

In the 1980s and 1990s, operating experience at NPPs revealed a number of weaknesses in the initial design, qualification, testing, and performance of motor-operated valves (MOVs). For example, some engineering analyses used in the original sizing and setting of MOVs were found to predict inadequately the thrust and torque required to open and close valves under design-basis conditions. Similarly, some testing methods used to measure valve stroke times under zero differential-pressure and flow conditions were found not to detect deficiencies that could prevent MOVs from performing their safety functions under design-basis conditions. Both regulatory and industry research programs later confirmed weaknesses in the performance of MOVs. Such programs included extensive NRC research to study the performance of MOVs under various flow, temperature, and voltage conditions, and a nuclear-industry-sponsored program by the Electric Power Research Institute (EPRI) to develop a computer methodology to predict the performance of MOVs under a wide range of operating conditions.

Responding to weaknesses found in the initial design, qualification, testing, and performance of MOVs, the NRC issued GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” (Ref. ~~1113~~12) in June 1989, which requested licensees to (1) ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases; (2) verify MOV switch settings initially and periodically; (3) test MOVs under design-basis conditions when practicable; (4) improve evaluations of MOV failures and necessary corrective action; and (5) trend MOV problems. The NRC staff evaluated various MOV NPP programs through onsite inspections of the design-basis capability of safety-related MOVs.

In support of the regulatory activities to ensure MOV design-basis capability, the NRC conducted a research program to test several MOVs under normal flow and blowdown conditions. The NRC summarized the results of this MOV research program in Information Notice (IN) 90-40, “Results of NRC-Sponsored Testing of Motor-Operated Valves,” (Ref. ~~1214~~13) dated June 5, 1990. The testing revealed that (1) more thrust was required to operate gate valves than predicted by standard industry methods; (2) some valves were internally damaged under blowdown conditions and their operating requirements were unpredictable; (3) static and low-flow testing might not predict valve performance under design-basis flow conditions; (4) during valve opening strokes, the highest thrust requirements might occur at unseating or in the flow stream; (5) partial valve stroking did not reveal the total thrust required to operate the valve; (6) torque, thrust, and motor operating parameters were needed to fully characterize MOV performance; and (7) reliable use of MOV diagnostic data requires accurate equipment and trained personnel.

To assist NPP licensees in responding to GL 89-10, the EPRI developed the MOV Performance Prediction Methodology (PPM) to determine dynamic thrust and torque requirements for gate, globe, and butterfly valves based on first principles of MOV design and operation. The EPRI described the methodology in Topical Report TR-103237, Revision 2, “EPRI MOV Performance Prediction Program,” (Ref. ~~13~~14~~5~~14) issued in April 1997. The EPRI MOV PPM program included the development of improved methods for the prediction and evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor-actuator rate-of-loading effects (load-sensitive behavior). The EPRI conducted numerous valve tests to provide data for the development and validation of the valve performance models and methods, including flow loop testing, parametric flow loop testing of butterfly valve disk designs, and in situ MOV testing. The NRC staff issued a safety evaluation (SE) in March 1996 (Ref. ~~1416~~15) accepting the EPRI MOV PPM with certain conditions and limitations. The NRC staff also issued supplements to the SE in February 1997 (Ref. ~~1517~~16), in April 2001 (Ref. ~~1618~~17~~), and~~17), ~~-~~in September 2002 ~~-~~(Ref. ~~1719~~18) and in February 2009 (Ref. 19) to address updates to the EPRI MOV PPM. The NRC staff alerted licensees to lessons learned from the EPRI MOV program in IN 96-48, “Motor-Operated Valve Performance Issues,” (Ref. ~~18~~20) dated August 21, 1996.

In September 1996, the NRC issued GL 96-05, "Periodic Verification of Design-Basis Capability `of Safety-Related Motor-Operated Valves," (Ref. ~~19~~21) to provide recommendations for ensuring the capability of safety-related MOVs to perform their design-basis functions over the long term. In response to GL 96-05, the NPP owners' groups developed an industrywide Joint Owners' Group (JOG) program on MOV periodic verification to obtain benefits from sharing information among licensees on MOV performance. Following an interim MOV program and extensive dynamic MOV testing at NPPs, in February 2004, the JOG submitted to the NRC the Topical Report MPR-2524, Revision 0, "Joint Owners' Group Motor Operated Valve Periodic Verification Program Summary," (Ref. ~~20~~22) providing long-term recommendations for the periodic verification of MOVs to be implemented by licensees as part of their commitments to GL 96-05. The NRC staff completed an SE on its evaluation of the JOG topical report in September 2006 (Ref. ~~21~~23)

In the late 1990s, the NRC conducted research to study the performance of alternating current (ac)-powered MOV motor actuators manufactured by Limitorque Corporation under various temperature and voltage conditions. For the Limitorque ac-powered motor-actuator combinations tested, the research indicated that (1) actuator efficiency might not be maintained at the "run" efficiency published by the manufacturer; (2) degraded voltage effects can be more severe than predicted by the square of the ratio of actual to rated motor voltage; (3) some motors produce more torque output than predicted by their nameplate rating; and (4) temperature effects on motor performance appeared consistent with the Limitorque guidance. The NRC documented its study of ac-powered MOV output in NUREG/CR-6478, "Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing," (Ref. ~~22~~24) issued in July 1997. The nuclear industry also evaluated the output capability of ac-powered MOVs at several plants. In response to the new information on ac-powered MOV performance, Limitorque provided updated guidance in its Technical Update 98-01, "Actuator Output Torque Calculation," (Ref. ~~23~~25) issued in May 1998, and Supplement 1, issued in July 1998, for the prediction of ac-powered MOV motor actuator output. The NRC alerted licensees to the new information on ac-powered MOV output in Supplement 1 to IN 96-48 (Ref. ~~24~~26), dated July 24, 1998.

Following the NRC review of ac-powered MOV performance, the NRC conducted research to study the performance of Limitorque direct current (dc)-powered MOV motor actuators under various temperature and voltage conditions. For the Limitorque dc-powered motor-actuator combinations tested, the research indicated that (1) the ambient temperature effects were more significant than predicted; (2) the use of a linear voltage factor needs to consider reduced speed, increased motor temperature, and reduced motor output; (3) the stroke-time increase is significant for some dc-powered MOVs under loaded conditions; and (4) the actuator efficiency may fall below the published "pullout" efficiency at low speed and high load conditions. The NRC documented this research in NUREG/CR-6620, "Testing of DC-Powered Actuators for Motor-Operated Valves," (Ref. ~~25~~27) issued in May 1999. In June 2000, the Boiling Water Reactor Owners' Group forwarded to the NRC the Topical Report NEDC-32958, "BWR Owners' Group DC Motor Performance Methodology—Predicting Capability and Stroke Time in DC Motor-Operated Valves," (Ref. ~~26~~28) issued in March 2000. In August 2001, the NRC issued Regulatory Issue Summary 2001-15, "Performance of DC-Powered Motor-Operated Valve Actuators," (Ref. ~~27~~29) to inform licensees of the availability of improved industry guidance for predicting dc-powered MOV actuator performance.

Through an extensive effort spanning over many years, the ASME QME Standards Committee revised Section QV in ASME QME-1 to incorporate the lessons learned from the MOV operating experience and research programs for the functional qualification of all power-operated valves. ~~The NRC staff participated in the activities of the ASME QME Standards Committee and its subcommittees during~~

~~the revision of ASME QME-1.~~ The staff presents its regulatory positions on ASME QME-1-2007 in Section C of this regulatory guide.

C. REGULATORY POSITION

1. Seismic Qualification of Electric and Active Mechanical Equipment

1.1 <u>Regulatory Positions on IEEE Std 344-2004</u>

1.1.1 *General NRC Staff Positions*

The IEEE Std 344-2004 is, in general, acceptable to the NRC staff for the seismic qualification of (1) electric equipment in new NPPs and (2) new addition or replacement electric equipment in operating NPPs, subject to the following provisions:

a. Rigorous seismic qualification by analysis, testing, or combined analysis and testing, as described in Clauses 7, 8, and 9 of the IEEE Std 344-2004, are acceptable methods for seismic qualification of electric equipment.

b. The use of experience data (earthquake or test experience data) for seismic qualification of electric equipment is subject to review by the NRC staff in areas such as (1) the credibility and completeness of the compilation of the experience database; (2) the inclusion rules and exclusion rules (termed “prohibited features” in IEEE Std 344-2004) for electric equipment in the experience database; (3) the justification used to demonstrate similarity among the member items in a reference equipment class; (4) the justification used to demonstrate ~~and the~~ similarity between electric equipment in the experience database and those in the NPP to be seismically qualified; ~~(4~~and (~~4~~5) the justification used to demonstrate the ~~reference~~ functionality of candidate equipment and the member items in a reference equipment class ~~functionality~~ during and after the earthquake~~; and (5) the credibility of similarity among member items of a reference equipment class if a generic reference equipment class is proposed~~;~~. As described in Clause 9.3 of IEEE Std 344-2004, similarity should include similarity in physical, functional, and dynamic characteristics between electric equipment in the experience database and those in the NPP to be seismically qualified, as well as; and (6) the justification used to demonstrate the similarity between seismic excitation noted in the experience database and the required seismic excitation for the electric equipment in the NPP to be seismically qualified~~.

c. The NRC staff concurs with the limitations given in IEEE 344-2004 Section 10.4.2 ~~does not generally find it acceptable to~~ for use of experience data (earthquake or test experience data) for seismic qualification for (1) certain active electric components that may inadvertently change state or chatter during an earthquake such that they do not consistently perform their intended safety functions during and/or after an earthquake, (e.g., ~~such as~~ certain types of relays, contactors, circuit breakers, switches, sensors, microprocessors-based components and potentiometers.)~~; (2) fragile electronic components, such as solid-state relays and microprocessors-based components; and (3) electric equipment, such as battery chargers, inverters, relay and control panels, switchgear, and motor control centers. The electric equipment mentioned in (3) above generally consists of enclosures constructed of metal frames with metal panels that contain some of the components described in (1) and (2) above, and the response and performance of these components (and therefore the performance of the equipment) under earthquake loadings in general are sensitive to their locations and the type of mountings in such equipment.~~ A seismic test may be needed to confirm that a component is not sensitive to high-frequency ground motion, if applicable.

d. If the licensee proposes to use test experience data to perform seismic qualification in accordance with IEEE-344-2004 Clause 10.3, then the licensee should submit for staff review and approval the details of the test experience database, including applicable implementation procedures, to assure the structural integrity and functionality of the in-scope electric equipment. Supporting documentation for equipment identified in the database should confirm that such equipment will remain functional during and after the equivalent effect of five postulated occurrences of OBE and one SSE in combination with other relevant static and dynamic loads consistent with the licensing basis for the facility.

~~e. For electric equipment exposed to harsh environments, aging, and earthquakes, the NRC staff does not find it acceptable to use experience data (earthquake or test experience data) for seismic qualification of equipment. The test sample shall be subjected to simulated OBE and SSE seismic vibrations in accordance with IEEE Std 344-2004.~~

~~f~~e. The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be consistent with the RRS of the specific plant equipment. ~~The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be continued beyond 33 Hz, in accordance with the RRS of a specific plant.~~

~~g.~~f. For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. ~~This guide refers to this phenomenon as the high-frequency ground motion concern.~~ As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. The vast majority of prior existing seismic qualification tests used input frequencies up to only 33 Hz. The use of these prior testing results should be justified by demonstrating that the frequency content of the power spectral density (PSD) of the test waveform is sufficient, in accordance with Annex B of IEEE 344-2004. ~~For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. This guide refers to this phenomenon as the high-frequency ground motion concern. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. For operating BWR plants, the seismic qualifications of some safety-related electric equipment were performed using IEEE-344-type tests with intentional high-frequency contents to account for concurrent BWR hydrodynamic loads. However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. These past test experience data are therefore not acceptable for the seismic qualification of high-frequency-sensitive equipment or fragile components. Furthermore, credit should not be taken for the inadvertent high frequencies present in some of the IEEE-344-type seismic qualification tests of equipment in the past, which may have shown the ZPA of the TRS to be up to 100 Hz. Ball joints and kinematics linkages of the shake tables could have generated these inadvertent high frequencies, and the NRC staff considers them to be noise signals that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies.~~

~~h~~

g. If new seismic qualification tests are planned for equipment in plants with the high-frequency ground motion~~ concern~~, ~~the test input waveforms should be properly formulated to address this concern. In order for the TRS to be valid for such plants,~~ the adequacy of the frequency content and the stationarity of the frequency content of the synthesized input waveforms used for the tests should be demonstrated. The frequency content of the Fourier transform of the test waveform or the frequency content of the power spectral density of the test waveform must be compatible with the amplified portion of the RRS. The Annex B, "Frequency Content and Stationarity," to the IEEE 344-2004 provides acceptable guidelines on frequency content and stationarity.

~~i~~h. For NPPs that were licensed with the elimination of the OBE, electric equipment should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events in accordance with Annex D of IEEE 344-2004 when followed by one full SSE (SECY-93-087) (Ref. 30) even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based on the OBE level in accordance with the licensing basis.~~Electric equipment should be qualified with five one-half SSE events followed by one full SSE event (SECY-93-087) (Ref. 28) even if the OBE of a plant is defined to be one-third of SSE or less. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events may be used in accordance with Annex D, "Test Duration and Number of Cycles," to IEEE Std 344-2004, when followed by one full SSE.~~

~~j~~i. The damping values used in analysis should be in accordance with ~~-~~the damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 31) issued in March 2007, or as approved in the plant licensing basis.~~.~~ Damping values other than those provided in the plant licensing/design basis or RG 1.61 may be used, subjected to staff review and approval, if documented test data supports the higher values.

~~31 for analysisfor analysis The IEEE Std 344-2004 recommended no damping values. The damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 29) issued in March 2007, are recommended. These damping values are the updated values currently acceptable to the NRC staff.~~

1.1.2 *Specific NRC Staff Positions*

The following are specific NRC staff positions, including exceptions and clarifications, on IEEE Standard~~d~~ 344-2004:

a. Clause 10.2.3.1 and Clause 10.3.3.1 (Experience—Attributes of Equipment Class)

The NRC staff will review, in detail, the attributes of the equipment for establishing the inclusion rules that constitute the earthquake or test experience reference equipment class as described in Clause 10.2.3.1 or Clause 10.3.3.1, respectively, to determine the acceptability of similarity arguments to define a reference equipment class.

To avoid addressing fatigue failure at low-cycle loads from earthquakes by simply considering it as a prohibited feature (Clause 10.2.3.1(b)) does not demonstrate successful equipment functionality under OBEs as required by the NRC regulations delineated in 10 CFR Part 100 (Ref. ~~30~~32), Appendix S to 10 CFR Part 50, or 10 CFR Part 52. Earthquake experience data or test data are needed to demonstrate that all electric equipment in the reference equipment class, including the enclosed or attached devices or subassemblies, performed successfully (structural integrity and specified functionality) under the equivalent of five OBE and one SSE loadings.

b. Clause 10.2.3.3 (Experience—Reference Equipment Class Functionality)

Detailed information about the justification used to demonstrate the reference equipment class functionality during and after the earthquake should be submitted for NRC review and approval.

c. Clause 10.2.4 (Earthquake Experience Data—Qualification of Candidate Equipment)

In-structure response spectra used as the RRS for the qualification of candidate equipment should be in accordance with the licensing/design basis or the Standard Review Plan, (Ref 33), Section 3.7.2, as applicable. The use of RRS other than those described in the licensing/design basis should be submitted for NRC staff review and approval.
~~design 3~~
~~The use of a median-centered horizontal in-structure response spectrum as the RRS for the candidate equipment is not acceptable. In-structure response spectra should be developed in accordance with the licensing basis and NRC guidance described in the latest revision of Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components." (Ref. 31)~~

d. Clause 10.3.2 (Test Experience Data—Test Experience Spectra (TES))

i. The use of the frequency-by-frequency mean of the successful TRS ~~is~~ may not be adequate to define TES. ~~When using test experience data, an equipment capacity factor has to be considered to obtain an equivalent confidence level for performance and to cover the uncertainties in high-level testing for an equipment class. The acceptable equipment capacity factor is 1.4 for TES (Refs. 32 and 33)..~~

Therefore, the NRC takes exception to the existing second sentence in the first paragraph of Clause 10.3.2. Instead, the following is acceptable to the NRC:

> The TES shall be the frequency-by-frequency mean of the response spectra from successful tests without malfunction. When using test experience data, both the mean and the standard deviation of the data leading to the TES curve should be provided for review and approval.
> ~~The TES shall be the frequency-by-frequency mean divided by 1.4 of the response spectra from successful tests without malfunction.~~

ii. The second paragraph of Clause 10.3.2 is not appropriate. The position acceptable to the NRC staff is the one stated in Section C.1.1.1.h~~i~~ of this regulatory guide.

e. Clause 10.3.3 (Test Experience Data—Characterization of Reference Equipment Class)

Clause 10.3.3 cites an example that significant natural frequencies of the reference equipment class would lie within approximately one-third octave. This will not provide an adequate range of significant natural frequencies of the reference equipment in a class. Instead of one-third octave, one-sixth octave should be used.

f. Clause 10.3.3.2 (Test Experience Data—Number of Independent Items for Reference Equipment Class)

Justification should be provided to show the adequacy of using a minimum of five independent items to define a reference equipment class for test experience.

~~g. Clause 10.3.4 (Test Experience Data—Qualification of Candidate Equipment)~~

~~In-structure response spectra used as the RRS for the qualification of candidate equipment should be in accordance with the licensing/design basis. The use of RRS other than those described in~~

~~the licensing/design basis should be submitted for NRC staff review and approval. The use of median-centered horizontal in-structure response spectrum as the RRS for the candidate equipment is not acceptable because it does not satisfy the current staff position stated in SRP 3.7.2 relative to modeling parameters variation.. In-structure response spectra should be in accordance with NRC guidance described in the latest revision of Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components.~~

~~The use of a median-centered horizontal in-structure response spectrum as the RRS for the candidate equipment is not acceptable. In-structure response spectra should be developed in accordance with the licensing basis and NRC guidance described in the latest revision of Regulatory Guide 1.122.~~

~~h~~. Clause 10.4.2 (Special Considerations~~Experience~~—Limitations)

The list of limitations for the use of earthquake- or test-experience-based methods for seismic qualification of equipment as described in Clause 10.4.2 may not be complete. The list should be expanded to include additional limitations as a result of new findings from testing new equipment or new studies.

h~~i~~ Annex C (Fragility Testing)

An example of determining the fragility level to single-frequency transient excitation of the equipment is to subject it to any single-frequency excitation such as sine-beat motion. The frequency range of the test excitation should be continued beyond 33 Hz, in accordance with the RRS of a specific plant.

i~~j~~. Annex D (Test Duration and Number of Cycles)

Figure D.2 is used to determine the equivalent number of fatigue cycles for a given filtered frequency and duration. Currently, IEEE Std 344-2004 does not develop guidance beyond 40 Hz. Therefore, justification should be provided for applications beyond 40 Hz.

~~k. Annex E (Statistically Independent Motions)~~

~~A coherence function of less than 0.5 and an absolute value of the correlation coefficient function of less than 0.3 are not acceptable.~~

~~The NRC positions on the numerical values for the coherence function and the correlation coefficient function for defining statistically independent motions are the same as in Reference 34, particularly the following:~~

~~i. For the coherence function, numerical values ranging from 0.0 to a maximum of 0.3 and an average of approximately 0.2 are acceptable.~~
~~ii. An absolute value of less than 0.16 for the correlation coefficient function is acceptable.~~

1.2 Regulatory Positions on ASME QME-1-2007

1.2.1 General NRC Staff Positions

In the discussion of the seismic qualification of some active mechanical equipment, ASME QME-1-2007 references IEEE Std-344-1987 (as addressed in NRC Regulatory Guide 1.100, Revision 2) or Nonmandatory Appendix QR-A. Such references appear in several sections of ASME QME-1-2007, such as Section QP-6400 for pumps, and Sections QV-7450 and QV-7650 for valves. The NRC staff finds these acceptable if they are applied consistent with the NRC staff positions delineated in this regulatory guide (Revision 3) and other relevant NRC regulatory documents.

The NRC staff finds Nonmandatory Appendix QR-A to ASME QME-1-2007 acceptable, in general, for seismic qualification of (1) active mechanical equipment in new NPPs; and (2) new addition or replacement active mechanical equipment in operating NPPs. However, the NRC staff acknowledges the statement in Section QR-A7500 that the section on test-experience-based qualification is currently not available in ASME QME-1-2007 and will be added in a later issue of the standard. In addition, the NRC has the following regulatory positions, including exceptions and clarifications, on Nonmandatory Appendix QR-A of ASME QME-1-2007:

a. In endorsing the use of ASME QME-1-2007, the staff noticed that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, ~~and~~ QP-E~~, and QV-A~~; and Mandatory Appendix QV-~~1~~II and Nonmandatory Appendix QV-A). The staff position is that, ~~once the user~~if a licensee commits to the use of non-mandatory appendic~~x~~es appendices in ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, then~~–~~the criteria and procedures delineated in those non-mandatory appendices ~~then~~ become part of the requirements for its qualification program, unless ~~the~~ specific deviations are requested and justified.

b. Rigorous seismic qualification by analysis or testing, as described in Sections QR-A7100, ~~and~~ QR-A7200 and QR-A7300 of ASME QME-1-2007, is an acceptable method for seismic qualification of active mechanical equipment.

c. ~~The~~ ASME Class 1, 2, and 3 active mechanical equipment should meet the requirements in the ASME Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code), Section III (Ref. ~~35~~34). The NRC staff recommends that a future revision of ASME QME-1 add this position to (1) Section QR-6000, "Qualification Specification," as item (j); and (2) Section QR-A7440, "Qualification of Candidate Equipment," as item (g).

d. The NRC staff will review the use of earthquake experience data for seismic qualification of active mechanical equipment as described in Section QR-A7400 of ASME QME-1-2007 in areas such as (1) the credibility and completeness of the compilation of the experience database; (2) the inclusion rules and exclusion rules for active mechanical equipment in the experience database; (3) the justification used to demonstrate similarity among the member items in a reference equipment class; (4) the justification used to demonstrate the functionality of candidate equipment during and after the earthquake: ~~The NRC staff will review the use of earthquake experience data for seismic qualification of active mechanical equipment in areas such as (1) the credibility and completeness of the compilation of the experience database; (2) the inclusion rules and exclusion rules for active mechanical equipment in the experience database; (3) the justification used to demonstrate similarity among the member items in a reference equipment class and the similarity between active mechanical equipment in the experience database and those in the NPP to be seismically qualified; (4~~and (45 the justification used to demonstrate the functionality of candidate equipment and the member items in a reference equipment class during and after the earthquake ~~functionality of candidate equipment and the member items in a reference equipment class functionality of candidate equipment during and after the earthquake; and : .(5) the credibility of similarity among member items of a reference equipment class if a~~

~~generic reference equipment class is proposed; and (6) . As described in Section QR-A7300 of ASME QME-1-2007, similarity should include similarity in physical, functional, and dynamic characteristics between mechanical equipment in the experience database and those in the NPP to be seismically qualified, as well as similarity between seismic excitation documented in the experience database and the required seismic excitation for the mechanical equipment in the NPP to be seismically qualified~~.

e. If the licensee proposes to use test experience data for seismic qualification, the licensee should submit for staff review and approval the details of the test experience database, including applicable implementation procedures, to assure the structural integrity and functionality of the in-scope mechanical equipment. Supporting documentation for equipment identified in the database should confirm that such equipment will remain functional during and after the equivalent effect of five postulated occurrences of an OBE and one SSE in combination with other relevant static and dynamic loads consistent with the licensing basis.

f. The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be consistent with the RRS of specific plant equipment~~The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be continued beyond 33 Hz, in accordance with the RRS of a specific plant.~~

g. For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. ~~This guide refers to this phenomenon as the high-frequency ground motion concern.~~ As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. The vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. The use of these prior testing results should be justified by demonstrating that the frequency content of the power spectral density (PSD) of the test waveform is sufficient in accordance with Annex B of IEEE 344-2004.
~~For operating BWR plants, the seismic qualification of some safety-related active mechanical equipment were performed using IEEE-344-type tests with intentional high-frequency contents to account for concurrent BWR hydrodynamic loads. However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. These past test experience data are therefore not acceptable for the seismic qualification of high-frequency-sensitive equipment or fragile components. Furthermore, credit should not be taken for the inadvertent high frequencies present in some of the IEEE-344-type seismic qualification tests of equipment in the past, which may have shown the ZPA of the TRS to be up to 100 Hz. Ball joints and kinematics linkages of the shake tables could have generated these inadvertent high frequencies, and the NRC staff considers them to be noise signals that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies.~~

h. If new seismic qualification tests are planned for active mechanical equipment in plants with the high-frequency ground motion~~ concern~~, ~~the test input waveforms should be properly formulated to address this concern. For the TRS to be valid for such plants,~~ the adequacy of the frequency content and the stationarity of the frequency content of the synthesized input waveforms used for the tests should be demonstrated. The frequency content of the Fourier transform of the test waveform or the frequency content of the power spectral density of the test waveform should be compatible with the amplified portion of the RRS. The Annex B to IEEE 344-2004 provides guidelines on frequency content and stationarity.

~~i.~~

~~j~~i. ~~For active mechanical equipment exposed to harsh environments, aging, and earthquakes, the staff does not find it acceptable to use experience data (earthquake or test experience data) for~~

~~seismic qualification of equipment. The test sample shall be subjected to simulated OBE and SSE seismic vibrations in accordance with IEEE Std 344-2004.~~

~~j.~~ For NPPs that were licensed with the elimination of the OBE, active mechanical equipment should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events in accordance with Annex D of IEEE 344-2004 when followed by one full SSE (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based on the OBE level in accordance with the licensing basis.~~Active mechanical equipment should be qualified with five one-half SSE events followed by one full SSE event (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events may be used in accordance with Annex D to IEEE Std 344-2004 when followed by one full SSE.~~

1.2.2 *Specific NRC Staff Positions*

The following are specific NRC staff positions, including exceptions and clarifications, on ASME QME-1-2007:

a. Section QR-A6200 (Damping)

The damping values used in analysis should be in accordance with the damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 31) issued in March 2007, or as approved in the plant licensing basis. Damping values other than those provided in the plant licensing/design basis or RG 1.61 may be used, subjected to staff review and approval, if documented test data supports the higher values.~~Instead of the damping value listed in Table QR-A6210-1,the acceptable damping values for analysis are those in accordance with the plant licensing basis. For new reactors, the damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," issued in March 2007, are recommended for analysis. These damping values are the updated values currently acceptable to the NRC staff. Damping values higher than those provided in RG 1.61 or the plant licensing basis may be used, subject to staff review and approval, if documented test data support the higher values.~~
~~Instead of the damping values listed in Table QR-A6210-1, the damping values listed in Tables 1 to 6 of NRC Regulatory Guide 1.61, Revision 1, are recommended. These damping values are the updated values currently acceptable to the NRC staff.~~

b. Section QR-A6300 (Seismic Qualification Requirements—Required Response Spectrum)

The Section QR-A6300 states that "For in-line active mechanical equipment qualified in accordance with QR-A7400 (Earthquake Experience-Based Qualification), the RRS is typically the building filtered response spectrum at the distribution system support attachments to the building." The use of the building filtered response spectrum at the distribution system support attachments to the building as the RRS for the in-line equipment may not be adequate. The RRS for in-line active mechanical equipment should account for the potential motion amplification of the distribution system.

c. Section QR-A7331 (Qualification by Similarity—Excitation)

The Section QR-A7331 states that “a conservative composite excitation may be generated by extrapolations or interpolations of data whose parameters are not identical but are justifiable. Likewise, excitation whose spectral content are significantly different may be used to generate lower-level composite estimates, providing that an account is taken of possible multi-axis response or cross-axis coupling, or both.” The licensee should provide detailed information, justifying this statement, to the NRC staff for review and approval.

d. Section QR-A7421 (Earthquake Experience-Based Qualification—Attributes of Equipment Class)

The NRC staff will review, in detail, the attributes of the equipment for establishing the inclusion rules that constitute the earthquake experience reference equipment class as described in Section QR-A7421 to determine the acceptability of similarity arguments to define a reference equipment class.

Section QR-A7421 also states the following:

> “Prohibited features should include any attributes that would contribute to fatigue failure from low cycle loads. The rules of this section apply to active mechanical equipment that may undergo 5 OBE’s or aftershocks and one SSE resulting in 60 full range stress cycles during plant life. If a component contains items which could experience a fatigue failure from low cycle loads (less than 60 full range stress cycles), it shall be evaluated in accordance with Section QR-A6800.”

The NRC regulations delineated in 10 CFR Part 100, Appendix S to 10 CFR Part 50, and 10 CFR Part 52 require the demonstration of successful equipment functionality under OBEs.

e. Section QR-A7423 (Earthquake Experience-Based Qualification—~~(~~Functionality during Earthquake)

Detailed information about the justification used to demonstrate the reference equipment class functionality during and after the earthquake should be submitted for NRC review and approval.

f. Section QR-A7431 (Earthquake Experience-Based Qualification—Inherently Rugged Active Mechanical Equipment)

To justify the active mechanical equipment class as an “inherently rugged active mechanical equipment” class, the licensee should provide, for NRC review, information regarding the operational or shipping loads as compared to the expected seismic loads that the equipment could be subjected to, and the explicit design standards applied to this equipment class. Detailed information regarding the simplified and reduced rules, including the technical justification and data for characterizing the inherently rugged active mechanical equipment class and the procedure for defining the seismic capacity for this equipment class (i.e., the earthquake experience spectrum), should also be provided to the NRC staff for review.

g. Section QR-A7432 (Earthquake Experience-Based Qualification—Limitations)

The limitations for the use of an earthquake-experience-based method of seismic qualification of equipment, as described in Section QR-A7432, may not be a complete list. The list should be expanded to include additional limitations as a result of new findings from testing of new equipment or new studies.

h. Section QR-A7440 (Earthquake Experience-Based Qualification—Qualification of Candidate Equipment)

In-structure response spectra used as the RRS for the qualification of candidate equipment should be in accordance with the licensing/design basis or SRP 3.7.2, as applicable. The use of RRS other than those described in the licensing/design basis should be submitted for NRC staff review and approval. ~~design The use of a median-centered horizontal in-structure response spectrum as the RRS for the candidate equipment is not acceptable. In-structure response spectra should be developed in accordance with the licensing basis and NRC guidance described in the latest revision of Regulatory Guide 1.122.~~

i. Section QR-A8330 (Qualification Report—Earthquake Experience-Based Qualification Documentation)

All ASME Class 1, 2, and 3 active mechanical equipment should comply with the ASME Code, Section III, requirements. The NRC staff recommends adding the following item to a future revision of ASME QME-1:

(f) compliance with the ASME Code, Section III, requirements for ASME Class 1, 2, and 3 active mechanical equipment.

j. Attachment C to Nonmandatory Appendix QR-A (Qualification of Pumps and Valves Using Natural Earthquake Experience Data)

Attachment C to Nonmandatory Appendix QR-A is based on the guidelines developed by the SQUG for USI A-46 plants. The provisions in the SQUG guidelines rely heavily on earthquake experience data that the staff considered reasonable for verification of existing equipment seismic adequacy, and the qualification of new and replacement equipment in older vintage USI A-46 plants only. The NRC staff has not accepted these SQUG guidelines for the seismic qualification of equipment in plants other than USI A-46 plants. The provisions outlined in Section QR-A7400, including the NRC staff's positions noted in Sections C.1.2.1 and C.1.2.2 of this regulatory guide, are acceptable for the seismic qualification of active mechanical components. In addition, contrary to the provisions in Section QR-A7400, the introduction to Attachment C states that the data have not been developed to conclusively demonstrate that pumps and valves function properly during earthquakes. Therefore, Attachment C in its current form does not fully comply with the requirements in 10 CFR Part 100. Attachment C also contains an error in Section C-2. The equipment frequency restriction should be greater than 8 Hz instead of less than 8 Hz.

2. Functional Qualification of Active Mechanical Equipment

2.1 <u>Regulatory Positions on ASME QME-1-2007</u>

2.1.1 *General NRC Staff Positions*

In general, the NRC staff finds ASME QME-1-2007 acceptable for the functional qualification of (1) active mechanical equipment in new NPPs; and (2) new addition or replacement of active mechanical equipment in operating NPPs, subject to the following provisions:

a. Appendices

In endorsing the use of ASME QME-1-2007, the staff acknowledged that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, ~~and~~ QP-E~~, and QV-A~~; and Mandatory Appendix QV-~~1)~~I~~.~~ and Nonmandatory Appendix QV-A.) The staff position is that, if a licensee commits to the use of non-mandatory appendixes in ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, then the criteria and procedures delineated in those non-mandatory appendices become part of the requirements for its qualification program, unless specific deviations are requested and justified.~~The NRC staff's position is that, once the user commits to the use of ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, the criteria and procedures delineated in those appendices then become mandatory and are the requirements for its qualification program. Justification must be provided for any deviations, which will be subjected to NRC staff review and approval.~~

b. Nonmandatory Appendix QR-B

This appendix recommends a methodology and describes the documentation that should be available in a user's files to demonstrate the qualification of nonmetallic parts, materials, or lubricants. It addresses the steps for the user of the active mechanical equipment to follow to qualify and maintain qualification of the nonmetallic material that is part of the active mechanical equipment. The NRC staff considers Nonmandatory Appendix QR-B to provide a reasonable approach to the qualification of nonmetallic material in active mechanical equipment.

c. Sections QDR and QP

The NRC staff considers Sections QDR and QP to provide a reasonable approach to the qualification of dynamic restraints and active pump assemblies, respectively. These sections have not changed from those in ASME QME-1-2002, and they still adequately document the state of the art of the nuclear industry in the qualification of dynamic restraints and active pump assemblies.

d. Section QV

The revision to ASME QME-1 reflects valve performance information obtained from nuclear industry programs and NRC's research since the development of ASME QME-1 in the 1980s. With the active involvement of industry personnel and the NRC staff in the development of

ASME QME-1-2007, only a few NRC staff exceptions and clarifications are necessary in this guide for Section QV, as described in Section C.2.1.2 of this regulatory guide below.

2.1.2 *Specific NRC Staff Positions*

a. The definition of "valve assembly" in Section QV-4000, "Definitions," refers to power-operated valves. The NRC staff considers the power actuators for valve assemblies to include all types of power actuators, such as motor, pneumatic, hydraulic, solenoid, and other drivers.[3]

b. The Section QV-6000, "Qualification Specification," states that the owner or owner's designee is responsible for identifying the functional requirements for a valve assembly, and that these requirements shall be provided in a qualification specification prepared in accordance with Mandatory Appendix QV-~~1~~I. The NRC staff considers Mandatory Appendix QV-~~1~~ I to be a necessary part of the implementation of Section QV of ASME QME-1-2007. For example, Mandatory Appendix QV-~~1~~ I provides the definitions of QV Category A and B valve assemblies used in Section QV of ASME QME-1-2007.

3 The guidance in ASME QME-1-2007 may also be used where applicable in the qualification of manually operated valves.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this draft regulatory guide. No imposition or backfit is intended or approved in connection with its issuance.

The NRC has issued this draft guide to encourage public participation in its development. The NRC will consider all public comments received in development of the final guidance document. Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in the active guide will be used in evaluating compliance with the regulations as discussed in this guide for license applications, license amendment applications, and exemption requests.

REGULATORY ANALYSIS

1. Statement of the Problem

The NRC issued Revision 2 of Regulatory Guide 1.100 in June 1988 to endorse (with exceptions and clarifications) IEEE Std 344-1987 and to describe acceptable methods for complying with the NRC's regulations governing the seismic qualification of NPP electric (including I&C components) and active mechanical equipment that is important to safety.

In 1981, the NRC issued Revision 0 of Regulatory Guide 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants." With a few exceptions and clarifications this guide endorsed American National Standards Institute (ANSI) N278.1-1975, "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard." ANSI N278.1 was eventually replaced by ASME QME-1.

Since the issuance of Revision 2 of Regulatory Guide 1.100 in 1988, and Revision 0 of Regulatory Guide 1.148 in 1981, two consensus standards, IEEE Std 344-2004 and ASME QME-1-2007, have been revised and issued in 2005 and 2007, respectively. With some exceptions and clarifications, this regulatory guide (Revision 3) endorses these two consensus standards and describes methods that the NRC staff considers acceptable in the areas of seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment. This revision of the regulatory guide will also subsume Regulatory Guide 1.148. Regulatory Guide 1.148 is intended to be deleted when Revision 3 of Regulatory Guide 1.100 is approved.

2. Objective

The objective of the regulatory action is to update the NRC's guidance in the area of seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment and to do so in a manner such that all regulatory guidance will be in a single document.

3. Alternatives Approaches

The NRC staff considered the following alterative approaches:

- Do not revise either Regulatory Guide 1.100 or Regulatory Guide 1.148.
- Update Regulatory Guide 1.100 & Regulatory Guide 1.148 as separate documents
- Combine the two Regulatory Guides into 1 updated document.~~.~~

3.1 Alternative 1: Do Not Revise Either Regulatory Guide 1.100 or Regulatory Guide 1.148

Under this alternative, the NRC would not revise either Regulatory Guide 1.100 or Regulatory Guide 1.148 and licensees would continue to rely on the current versions (Revision 2 and Revision 0 respectively), which are~~is~~ based on technology developed in the 1980s. Since then several newer developments that have improved the engineering understanding have been incorporated in updated standards. This alternative is considered the baseline or "no-action" alternative.

3.2 Alternative 2: Update Regulatory Guide 1.100 and Regulatory Guide 1.148 as Separate Documents

Under this alternative, the NRC would update Regulatory Guide 1.100 and Regulatory Guide 1.148 to reference the two latest consensus standards (see above) and to describe the NRC's positions on these two consensus standards. The staff has identified the following potential benefits associated with adopting alternative 2:

- Licensees would have NRC guidance on the use of the latest IEEE and ASME consensus standards related to the seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment.
- Regulatory efficiency would be somewhat improved by reducing uncertainty on acceptable methods to follow and by encouraging consistency in the seismic qualification of electric and active mechanical equipment and the functional qualification of active mechanical equipment. However, there could potentially be confusion over which Regulatory Guide to use, depending on the type of equipment being qualified. ~~The NRC reviews would be facilitated because licensee submittals would be more predictable and consistent~~.
- ~~Both the NRC and the nuclear industry would realize cost savings. From the NRC's perspective, relative to the baseline, the agency will incur one-time minimal incremental cost to issue the revised regulatory guide. However, the NRC should also realize cost savings associated with the review of licensee submittals; the ongoing cost savings associated with these reviews should more than offset the one-time cost.~~
- The updated regulatory guides will clarify for the industry the NRC positions on the two standards. The NRC staff expects that industry will realize a net savings by the efficiencies (e.g., fewer follow-up questions and revisions) associated with each licensee submission.

3.3 Alternative 3: Update Regulatory Guide 1.100 and combine it with Regulatory Guide 1.148

Under this alternative, the NRC would update Regulatory Guide 1.100 to reference the two latest consensus standards (see above) and to describe the NRC's positions on these two consensus standards. The staff has identified the following potential benefits associated with adopting alternative 3:

- Licensees would have NRC guidance on the use of the latest IEEE and ASME consensus standards related to the seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment in a single document.
- Regulatory efficiency would be improved by reducing uncertainty on acceptable methods to follow and by encouraging consistency in the seismic qualification of electric and active mechanical equipment and the functional qualification of active mechanical equipment. The NRC reviews would be facilitated because licensee submittals would be more predictable and consistent.
- Both the NRC and the nuclear industry would realize cost savings. From the NRC's perspective, relative to the baseline, the agency will incur one-time minimal incremental cost to issue the revised regulatory guide. However, the NRC should also realize cost savings associated with the review of licensee submittals; the ongoing cost savings associated with these reviews should more than offset the one-time cost.
- The updated regulatory guide will clarify for the industry the NRC positions on the two standards. The NRC staff expects that industry will realize a net savings by the efficiencies (e.g., fewer follow-up questions and revisions) associated with each licensee submission.

4. Conclusion

Based on this regulatory analysis, the staff recommends that the NRC revise Regulatory Guide 1.100, and combine the guidance of Regulatory Guide 1.148 into it~~.~~ The staff concludes that the proposed action will clarify the NRC positions on the two standards related to seismic qualification of electric and active mechanical equipment and functional qualification of active mechanical equipment. The staff also concludes that no adverse effects are associated with revising Regulatory Guide 1.100. Regulatory Guide 1.148 could be withdrawn once Revision 3 of Regulatory Guide 1.100 is approved.

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2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.

3. Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, June 1988.[5]

4. IEEE Std 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc., New York, NY, January 1987.[6]

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~~5~~6. ASME QME-1-1994, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," American Society of Mechanical Engineers, New York, NY, June 1994.[7]

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[3] All NRC regulations listed herein are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail PDR@nrc.gov.

[5] All regulatory guides listed herein were published by the U.S. Nuclear Regulatory Commission. Most are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. Active guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, VA 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax at (703) 605-6900. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555-0001; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548, and e-mail PDR@nrc.gov.

[6] Copies of Institute of Electrical and Electronics Engineers (IEEE) standards may be purchased from IEEE Service Center, 445 Hoes Lane, Piscataway, NJ 08854; telephone (800) 678-4333. Purchase information is available through the IEEE Web-based store at http://www.ieee.org.

[6] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Three Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.

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8 All generic letters (GLs) listed herein were published by the U.S. Nuclear Regulatory Commission and are available electronically through the Electronic Reading Room on the NRC’s public Web site, at http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/. Copies are also available for inspection or copying for a fee from the NRC’s Public Document Room at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555-0001; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail PDR@nrc.gov.

~~128~~ ~~The report can be accessed by the public from the Internet Web site http://www.osti.gov/bridge/servlets/purl/101081-bmPSJ9/webviewable/101081.pdf~~

~~10~~ 810 All information notices (INs) listed herein were published by the U.S. Nuclear Regulatory Commission and are available electronically through the Electronic Reading Room on the NRC’s public Web site, at http://www.nrc.gov/reading-rm/doc-collections/gen-comm/info-notices/. Copies are also available for inspection or copying for a fee from the NRC’s Public Document Room at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail PDR@nrc.gov

~~11~~911 Copies of the listed Electric Power Research Institute (EPRI) standards and reports may be purchased from EPRI, 3420 Hillview Ave., Palo Alto, CA 94304; telephone (800) 313-3774; fax (925) 609-1310.

~~14~~1615. Safety Evaluation, "Safety Evaluation on EPRI MOV Performance Prediction Methodology," U.S. Nuclear Regulatory Commission, Washington, DC, March 1996.

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~~[12] The report can be accessed by the public from the Internet Web site http://www.osti.gov/bridge/servlets/purl/101081-bmPSJ9/webviewable/101081.pdf~~

~~[13] Copies of American Society of Civil Engineers (ASCE) standards may be purchased from ASCE Publications, 1801 Alexander Bell Drive, Reston, VA 20191; telephone (800) 548-2723. Purchase information is available through the ASCE Web-based store at http://pubs.asce.org/books/.~~

34. ~~35. ——~~ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY~~, 2007~~.

~~Regulatory Guide 1.148 "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants" U. S. Nuclear Regulatory Commission, Washington, DC, March 1981.~~

U.S. NUCLEAR REGULATORY COMMISSION

June 2009
Revision 3

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.100

(Draft was issued as DG-1175, dated May 2008)

SEISMIC QUALIFICATION OF ELECTRICAL AND ACTIVE MECHANICAL EQUIPMENT AND FUNCTIONAL QUALIFICATION OF ACTIVE MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

This guide describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in the seismic qualification of electrical and active mechanical equipment and the functional qualification of active mechanical equipment for nuclear power plants (NPPs).

The general requirements for the seismic qualification of electrical and active mechanical equipment appear in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), and 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants" (Ref. 2). Particular sections include General Design Criterion (GDC) 1, "Quality Standards and Records"; GDC 2, "Design Bases for Protection Against Natural Phenomena"; and GDC 4, "Environmental and Dynamic Effects Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50; Criterion III, "Design Control"; Criterion XI, "Test Control"; and Criterion XVII, "Quality Assurance Records," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing

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The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public.

Regulatory guides are issued in 10 broad divisions—1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

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Plants," to 10 CFR Part 50; and Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50.

Section III, "Definitions," of Appendix S to 10 CFR Part 50 states that the structures, systems, and components (SSCs) required to withstand the effects of the safe-shutdown earthquake (SSE) ground motion or surface deformation are those necessary to assure (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1).
Section IV(a)(1)(ii) of Appendix S to 10 CFR Part 50 requires the NPP to be designed so that, if the SSE ground motion occurs, certain SSCs will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, the design of these safety-related SSCs must take into account applicable concurrent normal operating, functional, and accident-induced loads.
Section IV(a)(1)(iii) of Appendix S to 10 CFR Part 50 requires the safety functions of SSCs to be assured during and after the vibratory ground motion associated with the SSE ground motion through design, testing, or qualification methods.[1]

The general requirements for the functional qualification of active mechanical equipment also appear in 10 CFR Part 50 and 10 CFR Part 52. In 10 CFR Part 50, particular sections include GDC 1, GDC 4, GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 30, "Quality of Reactor Pressure Boundary," GDC 37, "Testing of Emergency Core Cooling System," GDC 40, "Testing of Containment Heat Removal System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," GDC 46, "Testing of Cooling Water System," and GDC 54, "Systems Penetrating Containment," of Appendix A to 10 CFR Part 50, as well as Criteria III, XI, and XVII of Appendix B to 10 CFR Part 50.

This regulatory guide contains information collection requirements covered by 10 CFR Part 50 and 10 CFR Part 52 that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0151. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays currently valid OMB control numbers.

[1] Appendix S to 10 CFR Part 50 applies to applicants for a design certification or combined license pursuant to 10 CFR Part 52 or a construction permit or operating license pursuant to 10 CFR Part 50 after January 10, 1997. However, the earthquake engineering criteria in Section VI, "Application to Engineering Design," of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria" (Ref. 3), continue to apply to either an operating license applicant or holder with a construction permit issued before January 10, 1997.

B. DISCUSSION

Background

The NRC issued Revision 2 of Regulatory Guide (RG) 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants" (Ref. 4), in June 1988. With a few exceptions and clarifications, it endorsed the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" (Ref. 5), issued January 1987, and extended the application of that standard to the seismic qualification of mechanical equipment. In extending the application of IEEE Std. 344-1987 to mechanical equipment, the NRC staff recognized differences in seismic qualification methods for electrical equipment[2] (including instrumentation and control (I&C) components) and mechanical equipment. Specifically, Revision 2 of this regulatory guide stated that the seismic qualification of mechanical equipment by analysis is permitted when such equipment can be modeled to adequately predict its response. Revision 2 also stated that the American Society of Mechanical Engineers (ASME) was developing a standard for the seismic qualification of mechanical equipment and, upon its publication; the NRC staff would review it for suitability for endorsement by a revision of this regulatory guide.

In March 1981, the NRC issued Revision 0 of RG 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants" (Ref. 6). With a few exceptions and clarifications, this guide endorsed American National Standards Institute (ANSI) N278.1-1975, "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard" (Ref. 7).

In 1994, ASME issued a standard, ASME QME-1-1994, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants" (Ref. 8). This ASME standard eventually replaced the ANSI N278.1 standard. The ASME QME-1 standard covers both the seismic qualification and the functional qualification of active mechanical equipment. ASME subsequently revised and reissued the standard in 1997, 2000, and 2002, with the last revision issued in November 2007 as ASME QME-1-2007 (Ref. 9). Furthermore, the IEEE updated IEEE Std. 344-1987 and issued it as IEEE Std. 344-2004 (Ref. 10) in June 2005.

The NRC is revising this RG (i.e., Revision 3 of RG 1.100) to endorse, with exceptions and clarifications, IEEE Std. 344-2004 and ASME QME-1-2007. (This is the first time the NRC has endorsed ASME QME-1.) This revision of the RG will also subsume RG 1.148.   Specifically, Sections B.1 and C.1 of this RG endorse, with exceptions and clarifications, the entire IEEE Std. 344-2004 and Section QR, "General Requirements," and Nonmandatory Appendix QR-A, "Seismic Qualification of Active Mechanical Equipment," to ASME QME-1-2007 for the seismic qualification of electrical and active mechanical equipment. Sections B.2 and C.2 of this RG endorse, with exceptions and clarifications, Section QR and the remaining sections of ASME QME-1-2007 (except Nonmandatory Appendix QR-A) for the functional qualification of active mechanical equipment. The ASME QME-1 standard defines active mechanical equipment as "Mechanical equipment containing moving parts, which, in order to accomplish its required function as defined in the Qualification Specification, must undergo or

[2] Hereafter in this RG, the term "electrical equipment" means an assembly of electrical and electronic components designed and manufactured to perform specific functions, and the term "electrical component" or "electronic component" means items from which the equipment is assembled (e.g., resistors, capacitors, wires, connectors, microprocessors, switches, springs, and instrumentation and control items).

prevent mechanical movement. This includes any internal components or appurtenances whose failure degrades the required function of the equipment."

1. Seismic Qualification of Electrical and Active Mechanical Equipment

The major change from IEEE Std. 344-1987 to IEEE Std. 344-2004 is the update and expansion of Clause 10, "Experience," which describes the use of experience data as a method for seismic qualification of Class 1E electrical equipment (including I&C components). Experience data include earthquake experience data and test experience data. Nonmandatory Appendix QR-A to ASME QME-1-2007, which has been updated and expanded from Nonmandatory Appendix QR-A to ASME QME-1-2002, also includes the use of experience data as a method for the seismic qualification of active mechanical equipment.

The use of earthquake experience data for the seismic qualification of electrical and mechanical equipment has its origin in the NRC research program associated with Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Mechanical and Electrical Equipment in Operating Nuclear Power Plants." In 1980, the NRC staff raised a safety concern that licensees had not conducted the seismic qualification of electrical and mechanical equipment in some older NPPs (i.e., plants with construction permit applications docketed before about 1972), in accordance with the licensing criteria for the seismic qualification of equipment acceptable at that time (i.e., IEEE Std. 344-1975 (Ref. 11) and RG 1.100, Revision 1 (Ref. 12), issued August 1977). Therefore, equipment in the older NPPs may not have been adequately qualified to ensure its structural integrity or proper functionality in the event of an SSE ground motion. As a result, the NRC established the USI A-46 program in December 1980 and, in February 1987, issued Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46" (Ref. 13), to address this safety concern. The NRC staff categorized approximately 70 NPP units in the United States as "USI A-46 plants."

In 1982, the Seismic Qualification Utility Group (SQUG) developed a database using earthquake experience and test experience to address USI A-46. Because of the scarcity of data on equipment that was subjected to strong earthquake motion in U.S. NPPs, the SQUG and its contractors performed a pilot study to determine the feasibility of using actual earthquake experience data from nonnuclear plants located worldwide (e.g., fossil-fueled power plants, substations, petrochemical plants) and existing test experience data from domestic NPPs to evaluate the performance of electrical and mechanical equipment in those facilities to infer the susceptibility of similar NPP equipment to seismic loads. The SQUG concluded, and the NRC agreed, that the use of experience data was feasible for the purpose of verifying the seismic adequacy of equipment in the older, USI A-46 plants. The staff does not accept the use of SQUG guidelines for the seismic qualification of equipment in non-USI A-46 plants licensed under 10 CFR Part 50 or in plants licensed under 10 CFR Part 52.

Large uncertainties exist in the seismic qualification of equipment, as a class, on the basis of earthquake experience data, because (1) it is difficult to compile a credible earthquake experience database (e.g., estimation of ground and floor earthquake excitations used in the earthquake experience database), (2) the inclusion and exclusion rules (termed "prohibited features" in IEEE Std. 344-2004) of equipment in the database may not be complete, (3) the similarity between equipment in fossil or petrochemical plants in the database and the equipment in NPPs is difficult to establish; and, most importantly, (4) generally, there is not sufficient credible information from the earthquake experience database to provide assurance that certain active electrical equipment will function properly during earthquakes.

In using the test experience data for the seismic qualification of electrical equipment, quantifying the damage potential of equipment under testing should capture the combination of input motion and the equipment item exhibiting a particular malfunction. Given the likelihood that the resonant frequency for items of equipment of the same class may differ significantly, multiple malfunction mechanisms for components and subcomponents should be considered in comparing the test response spectra (TRS) and the required response spectra (RRS).

The technologies and designs of certain electrical components (such as certain types of relays and microprocessor-based components) have undergone significant changes since the NRC issued Revision 2 of this RG, as a result of the more prevalent use of digital rather than analog I&C components. Some solid-state relays and microprocessor-based components may be sensitive to earthquake excitations. The staff considers the use of test experience data from the older electrical components of this type to be inappropriate and unacceptable for the seismic qualification of the new generation of such electrical components. Furthermore, since no new NPPs have been built in the United States since the early 1980s, a number of manufacturers of electrical or active mechanical equipment are no longer in business, and the appropriateness of using the test experience of old equipment made by manufacturers no longer in business for the seismic qualification of modern equipment designs made by different manufacturers is highly questionable.

Recent studies related to applications for early site permits at certain hard-rock-based plants along the east coast of the United States indicated that the site-specific spectra may exceed the certified design spectra of those new plants in the high-frequency range (20 hertz (Hz) and above). This exceedance cannot always be eliminated, even with incoherency added to the soil-structure interaction analyses. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. For operating boiling-water reactor (BWR) plants, the seismic qualifications of some safety-related electrical and active mechanical equipment were performed using IEEE Std. 344-type tests with intentional high-frequency contents to account for concurrent BWR hydrodynamic loads. However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz, although the TRS may have shown a zero period acceleration (nonamplified frequency range) up to 100 Hz. Ball joints and kinematics linkages of the shake tables generated these inadvertent high frequencies that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies. Therefore, any attempt to use such past test experience data for the seismic qualification of high-frequency-sensitive equipment or components in such a plant is not appropriate unless the frequency content of the power spectral density (PSD) of the test waveform has been evaluated in accordance with Annex B, “Frequency Content and Stationarity,” to IEEE Std. 344-2004. When licensees plan new seismic qualification tests for equipment in such plants, the formulation of the test input waveforms should properly consider this high-frequency excitation.

2. Functional Qualification of Active Mechanical Equipment

The ASME QME-1-2007 standard describes requirements and guidelines for qualifying active mechanical equipment used in NPPs. The foreword to the standard indicates that it may be applied to future NPPs or existing operating NPP component replacements, modifications, or additions, as determined by regulators and the NPP licensees. The ASME QME-1-2007 provides functional qualification guidance for nonmetallic parts, dynamic restraints, pumps, and valves. The following sections and appendices of ASME QME-1-2007 provide the functional qualification guidance for this active mechanical equipment: (1) Section QR, (2) Nonmandatory Appendix QR-B, “Guide for Qualification of Nonmetallic Parts,” (3) Section QDR, “Qualification of Dynamic Restraints,” and its

Nonmandatory Appendices QDR-A, "Functional Specification for Dynamic Restraints," QDR-B, "Restraint Similarity," and QDR-C, "Typical Values of Restraint Functional Parameters," (4) Section QP, "Qualification of Active Pump Assemblies," and its Nonmandatory Appendices QP-A, "Pump Specification Checklist," QP-B, "Pump Shaft-Seal System Specification Checklist," QP-C, "Pump Turbine Driver Specification Checklist," QP-D, "Pump Similarity Checklist," and QP-E, "Guidelines for Shaft-Seal System Material and Design Consideration," (5) Section QV, "Functional Qualification Requirements for Active Valve Assemblies for Nuclear Power Plants," its Mandatory Appendix QV-I, "Qualification Specification for Active Valves," and its Nonmandatory Appendix QV-A, "Functional Specification for Active Valves for Nuclear Power Plants," and (6) Section QV-G, "Guide to Section QV: Determination of Valve Assembly Performance Characteristics." The major change from ASME QME-1-2002 to ASME QME-1-2007, in terms of the functional qualification of mechanical equipment, is a complete rewrite of Section QV and the new Mandatory Appendix QV-I.

In the 1980s and 1990s, operating experience at NPPs revealed a number of weaknesses in the initial design, qualification, testing, and performance of motor-operated valves (MOVs). For example, some engineering analyses used in the original sizing and setting of MOVs inadequately predicted the thrust and torque required to open and close valves under design-basis conditions. Similarly, some testing methods used to measure valve stroke times under zero differential-pressure and flow conditions did not detect deficiencies that could prevent MOVs from performing their safety functions under design-basis conditions. Both regulatory and industry research programs later confirmed weaknesses in the performance of MOVs. Such programs included extensive NRC research to study the performance of MOVs under various flow, temperature, and voltage conditions, and a nuclear-industry-sponsored program by the Electrical Power Research Institute (EPRI) to develop a computer methodology to predict the performance of MOVs under a wide range of operating conditions.

Responding to weaknesses found in the initial design, qualification, testing, and performance of MOVs, the NRC issued GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (Ref. 14), in June 1989, which requested licensees to (1) ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, (2) verify MOV switch settings initially and periodically, (3) test MOVs under design-basis conditions when practicable, (4) improve evaluations of MOV failures and necessary corrective action, and (5) trend MOV problems. The NRC staff evaluated various MOV NPP programs through onsite inspections of the design-basis capability of safety-related MOVs.

In support of the regulatory activities to ensure MOV design-basis capability, the NRC conducted a research program to test several MOVs under normal flow and blowdown conditions. The NRC summarized the results of this MOV research program in Information Notice (IN) 90-40, "Results of NRC-Sponsored Testing of Motor-Operated Valves" (Ref. 15), dated June 5, 1990. The tests revealed that (1) more thrust was required to operate gate valves than predicted by standard industry methods, (2) some valves were internally damaged under blowdown conditions and their operating requirements were unpredictable, (3) static and low-flow testing might not predict valve performance under design-basis flow conditions, (4) during valve opening strokes, the highest thrust requirements might occur at unseating or in the flow stream, (5) partial valve stroking did not reveal the total thrust required to operate the valve, (6) torque, thrust, and motor operating parameters were needed to fully characterize MOV performance, and (7) reliable use of MOV diagnostic data requires accurate equipment and trained personnel.

To assist NPP licensees in responding to GL 89-10, EPRI developed the MOV performance prediction methodology (PPM) to determine dynamic thrust and torque requirements for gate, globe, and butterfly valves, based on first principles of MOV design and operation. EPRI described the methodology in Topical Report (TR)-103237, Revision 2, "EPRI MOV Performance Prediction Program" (Ref. 16), issued April 1997. The EPRI program included the development of improved methods for the prediction and evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor-actuator rate-of-loading effects (load-sensitive behavior). EPRI conducted numerous valve tests to provide data for the development and validation of the valve performance models and methods, including flow loop testing, parametric flow loop testing of butterfly valve disk designs, and in situ MOV testing. The NRC staff issued a safety evaluation (SE) in March 1996, "Safety Evaluation on EPRI MOV Performance Prediction Methodology" (Ref. 17), accepting the EPRI program with certain conditions and limitations. The NRC staff also issued supplements to the SE in February 1997 (Ref. 18), April 2001 (Ref. 19), September 2002 (Ref.209), and February 2009 (Ref. 21), to address updates to the EPRI MOV program. The NRC staff alerted licensees to lessons learned from the EPRI program in IN 96-48, "Motor-Operated Valve Performance Issues" (Ref. 22), dated August 21, 1996.

In September 1996, the NRC issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Ref. 23), to provide recommendations for ensuring the capability of safety-related MOVs to perform their design-basis functions over the long term. In response to GL 96-05, the NPP owners groups developed an industry wide Joint Owners Group (JOG) program on MOV periodic verification to obtain benefits from sharing information among licensees on MOV performance. Following an interim MOV program and extensive dynamic MOV testing at NPPs, in February 2004, the JOG submitted to the NRC the Topical Report MPR-2524, Revision 0, "Joint Owners' Group Motor-Operated Valve Periodic Verification Program Summary" (Ref. 24), providing long-term recommendations for the periodic verification of MOVs to be implemented by licensees as part of their commitments to GL 96-05. The NRC staff issued its "Final Safety Evaluation on Joint Owners' Group Program on Motor-Operated Valve Periodic Verification" in September 2006 (Ref. 25).

In the late 1990s, the NRC conducted research to study the performance of alternating current (ac)-powered MOV motor actuators manufactured by Limitorque Corporation under various temperature and voltage conditions. For the Limitorque ac-powered motor-actuator combinations tested, the research indicated that (1) actuator efficiency might not be maintained at the "run" efficiency published by the manufacturer, (2) degraded voltage effects can be more severe than predicted by the square of the ratio of actual-to-rated motor voltage, (3) some motors produce more torque output than predicted by their nameplate rating, and (4) temperature effects on motor performance appeared consistent with the Limitorque guidance. The NRC documented its study of ac-powered MOV output in NUREG/CR-6478, "Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing" (Ref. 26), issued July 1997. The nuclear industry also evaluated the output capability of ac-powered MOVs at several plants. In response to the new information on ac-powered MOV performance, Limitorque provided updated guidance in its Technical Update 98-01, "Actuator Output Torque Calculation" (Ref. 27), issued May 1998, and Supplement 1, issued in July 1998, for the prediction of ac-powered MOV motor actuator output. The NRC alerted licensees to the new information on ac-powered MOV output in Supplement 1 to IN 96-48 (Ref. 28), dated July 24, 1998.

Following its review of ac-powered MOV performance, the NRC conducted research to study the performance of Limitorque direct current (dc)-powered MOV motor actuators under various temperature and voltage conditions. For the Limitorque dc-powered motor-actuator combinations tested, the research indicated that (1) the ambient temperature effects were more significant than predicted, (2) the use of a linear voltage factor needs to consider reduced speed, increased motor temperature, and reduced motor output, (3) the stroke-time increase is significant for some dc-powered MOVs under loaded conditions, and (4) the actuator efficiency may fall below the published "pullout" efficiency at low-speed and high-load conditions. The NRC documented this research in NUREG/CR-6620, “Testing of dc-Powered Actuators for Motor-Operated Valves” (Ref. 29), issued May 1999. In June 2000, the BWR Owners Group forwarded to the NRC its TR NEDC-32958, “BWR Owners Group dc Motor Performance Methodology—Predicting Capability and Stroke Time in dc Motor-Operated Valves” (Ref. 30), issued March 2000. In August 2001, the NRC issued Regulatory Issue Summary 2001-15, “Performance of dc-Powered Motor-Operated Valve Actuators” (Ref. 31), to inform licensees of the availability of improved industry guidance for predicting dc-powered MOV actuator performance.

Through an extensive effort spanning many years, the ASME QME Standards Committee revised Section QV in ASME QME-1 to incorporate the lessons learned from the MOV operating experience and research programs for the functional qualification of all power-operated valves. The staff presents its regulatory positions on ASME QME-1-2007 in Section C of this RG.

C. REGULATORY POSITION

1. Seismic Qualification of Electrical and Active Mechanical Equipment

1.1 Regulatory Positions on IEEE Std. 344-2004

1.1.1 General NRC Staff Positions

The IEEE Std. 344-2004 is, in general, acceptable to the NRC staff for the seismic qualification of (1) electrical equipment in new NPPs and (2) new or replacement electrical equipment in operating NPPs, subject to the following provisions:

a. Rigorous seismic qualification by analysis, testing, or combined analysis and testing, as described in Clauses 7, 8, and 9 of IEEE Std. 344-2004, are acceptable methods for the seismic qualification of electrical equipment.

b. The use of experience data (earthquake or test experience data) for the seismic qualification of electrical equipment is subject to review by the NRC staff in areas such as (1) the credibility and completeness of the compilation of the experience database, (2) the inclusion and exclusion rules (termed "prohibited features" in IEEE Std. 344-2004) for electrical equipment in the experience database, (3) the justification used to demonstrate the similarity among the member items in a reference equipment class, (4) the justification used to demonstrate the similarity between electrical equipment in the experience database and equipment in the NPP for seismic qualification purposes, and (5) the justification used to demonstrate the functionality of candidate equipment and the member items in a reference equipment class during and after an earthquake.

c. The NRC staff concurs with the limitations given in IEEE Std. 344-2004, Section 10.4.2, for the use of experience data (earthquake or test experience data) for the seismic qualification of certain active electrical components that may inadvertently change state or chatter during an earthquake and thus may not consistently perform their intended safety functions during and/or after an earthquake (e.g., certain types of relays, contactors, circuit breakers, switches, sensors, microprocessor-based components, and potentiometers). A seismic test may be needed to confirm that a component is not sensitive to high-frequency ground motion, if applicable.

d. If the licensee proposes to use test experience data for seismic qualification in accordance with IEEE Std. 344-2004, Clause 10.3, the licensee should submit, for staff review and approval, the details of the test experience database, including applicable implementation procedures, to ensure the structural integrity and functionality of the in-scope electrical equipment. Supporting documentation for equipment identified in the database should confirm that such equipment will remain functional during and after the equivalent effect of five postulated occurrences of an operating-basis earthquake (OBE) and one SSE, in combination with other relevant static and dynamic loads, consistent with the licensing basis for the facility.

e. The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be consistent with the RRS of the specific plant equipment.

f. For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. The vast majority of prior existing seismic qualification tests used input frequencies up to only 33 Hz. The use of these prior testing results should be justified by demonstrating that the frequency content of the PSD of the test waveform is sufficient, in accordance with Annex B to IEEE Std. 344-2004.

g. If licensees plan new seismic qualification tests for equipment in plants with high-frequency ground motion, the tests should demonstrate the adequacy of the frequency content and the stationarity of the frequency content of the synthesized input waveforms. The frequency content of the Fourier transform of the test waveform or the frequency content of the PSD of the test waveform must be compatible with the amplified portion of the RRS. Annex B to IEEE Std. 344-2004 provides acceptable guidelines on frequency content and stationarity.

h. For NPPs that were licensed with the elimination of the OBE, electrical equipment qualified by testing should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events, in accordance with Annex D to IEEE Std. 344-2004, when followed by one full SSE (SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," issued April 1993) (Ref. 32), even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based on the OBE level, in accordance with the licensing basis.

i. The damping values used in the analysis should be in accordance with the damping values listed in Table 6 of RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants" (Ref. 33), issued March 2007, or as approved in the plant licensing basis. Licensees may use damping values other than those provided in the plant licensing or design basis, or in RG 1.61, subject to staff review and approval, if documented test data support the higher values.

1.1.2 Specific NRC Staff Positions

The following are specific NRC staff positions, including exceptions and clarifications, on IEEE Std. 344-2004:

a. Clause 10.2.3.1 and Clause 10.3.3.1 (Experience—Attributes of Equipment Class)

The NRC staff will review, in detail, the attributes of the equipment for establishing the inclusion rules that constitute the reference equipment class for earthquake or test experience, as described in Clause 10.2.3.1 or Clause 10.3.3.1, respectively, to determine the acceptability of similarity arguments to define a reference equipment class.

To avoid addressing fatigue failure at low-cycle loads from earthquakes by simply considering it as a prohibited feature (Clause 10.2.3.1(b)) does not demonstrate successful equipment functionality under OBEs as required by the NRC regulations delineated in 10 CFR Part 100, Appendix S to 10 CFR Part 50, or 10 CFR Part 52. Earthquake experience data or test data are needed to demonstrate that all electrical equipment in the reference equipment class, including the enclosed or attached devices or subassemblies, performed successfully (structural integrity and specified functionality) under the equivalent of five OBE and one SSE loadings.

b. Clause 10.2.3.3 (Experience—Reference Equipment Class Functionality)

Licensees should submit, for NRC review and approval, detailed information on the justification used to demonstrate the reference equipment class functionality during and after an earthquake.

c. Clause 10.2.4 (Earthquake Experience Data—Qualification of Candidate Equipment)

Licensees should ensure that in-structure response spectra used as the RRS for the qualification of candidate equipment is in accordance with the licensing or design basis or NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP)" (Ref. 34), Section 3.7.2, as applicable. They should submit, for NRC review and approval, information on the use of RRS other than that described in the licensing or design basis.

d. Clause 10.3.2 (Test Experience Data—Test Experience Spectra)

(1) The use of the frequency-by-frequency mean of the successful TRS may not be adequate to define test experience spectra (TES).

Therefore, the NRC takes exception to the existing second sentence in the first paragraph of Clause 10.3.2. Instead, the following is acceptable to the NRC:

> The TES shall be the frequency-by-frequency mean of the response spectra from successful tests without malfunction. When using test experience data, both the mean and the standard deviation of the data leading to the TES curve should be provided for review and approval.

(2) The second paragraph of Clause 10.3.2 is not appropriate. The position acceptable to the NRC staff is the one stated in Section C.1.1.1.h of this regulatory guide.

e. Clause 10.3.3 (Test Experience Data—Characterization of Reference Equipment Class)

Clause 10.3.3 cites an example that significant natural frequencies of the reference equipment class would lie within approximately a one-third octave. This will not provide an adequate range of significant natural frequencies of the reference equipment in a class. The example should use a one-sixth octave instead of a one-third octave.

f. Clause 10.3.3.2 (Test Experience Data—Number of Independent Items for Reference Equipment Class)

Licensees should provide justification to show the adequacy of using a minimum of five independent items to define a reference equipment class for test experience.

g. Clause 10.4.2 (Special Considerations—Limitations)

The list of limitations for the use of earthquake- or test-experience-based methods for the seismic qualification of equipment, as described in Clause 10.4.2, may not be complete. The list should be expanded to include additional limitations as a result of new findings from testing new equipment or new studies.

h Annex C (Fragility Testing)

An example of determining the fragility level to single-frequency transient excitation of the equipment is to subject it to any single-frequency excitation such as sine-beat motion. The frequency range of the test excitation should be continued beyond 33 Hz, in accordance with the RRS of a specific plant.

i. Annex D (Test Duration and Number of Cycles)

Figure D.2 is used to determine the equivalent number of fatigue cycles for a given filtered frequency and duration. Currently, IEEE Std. 344-2004 does not develop guidance beyond 40 Hz. Therefore, licensees should provide justification for applications beyond 40 Hz.

1.2 Regulatory Positions on ASME QME-1-2007

1.2.1 General NRC Staff Positions

In the discussion of the seismic qualification of some active mechanical equipment, ASME QME-1-2007 references IEEE Std. 344-1987 (as addressed in RG 1.100, Revision 2) or Nonmandatory Appendix QR-A. Such references appear in several sections of ASME QME-1-2007, such as Section QP-6400 for pumps and Sections QV-7450 and QV-7650 for valves. The NRC finds these acceptable if their application is consistent with the NRC staff positions delineated in this RG (Revision 3) and other relevant NRC regulatory documents.

The NRC staff finds Nonmandatory Appendix QR-A to ASME QME-1-2007 acceptable, in general, for the seismic qualification of (1) active mechanical equipment in new NPPs, and (2) new or replacement active mechanical equipment in operating NPPs. However, the NRC staff acknowledges the statement in Section QR-A7500 that the section on test-experience-based qualification is currently not available in ASME QME-1-2007 and will be added in a later issue of the standard. In addition, the NRC has the following regulatory positions, including exceptions and clarifications, on Nonmandatory Appendix QR-A to ASME QME-1-2007:

a. In endorsing the use of ASME QME-1-2007, the staff noticed that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, and QP-E; and Mandatory Appendix QV-I and Nonmandatory Appendix QV-A). The staff position is that, if a licensee commits to the use of nonmandatory appendices in ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, then the criteria and procedures delineated in those nonmandatory appendices become part of the requirements for its qualification program, unless specific deviations are requested and justified.

b. Rigorous seismic qualification by analysis or testing, as described in Sections QR-A7100, QR-A7200, and QR-A7300 of ASME QME-1-2007, is an acceptable method for the seismic qualification of active mechanical equipment.

c. ASME Class 1, 2, and 3 active mechanical equipment should meet the requirements in the ASME Boiler and Pressure Vessel Code (ASME Code), Section III (Ref. 35). The NRC staff recommends that a future revision of ASME QME-1 add this position to (1) Section QR-6000, “Qualification Specification,” as item (j), and (2) Section QR-A7440, “Qualification of Candidate Equipment,” as item (g).

d. The NRC staff will review the use of earthquake experience data for the seismic qualification of active mechanical equipment, as described in Section QR-A7400 of ASME QME-1-2007, in areas such as (1) the credibility and completeness of the compilation of the experience database, (2) the inclusion and exclusion rules for active mechanical equipment in the experience database, (3) the justification used to demonstrate the similarity among the member items in a reference equipment class, (4) the justification used to demonstrate the similarity between active mechanical equipment in the experience database and equipment in the NPP, for seismic qualification purposes, and (5) the justification used to demonstrate the functionality of candidate equipment and the member items in a reference equipment class during and after an earthquake.

e. If the licensee proposes to use test experience data for seismic qualification, the licensee should submit, for NRC review and approval, the details of the test experience database, including applicable implementation procedures, to ensure the structural integrity and functionality of the in-scope mechanical equipment. Supporting documentation for equipment identified in the database should confirm that such equipment will remain functional during and after the equivalent effect of five postulated occurrences of an OBE and one SSE, in combination with other relevant static and dynamic loads, consistent with the licensing basis.

f. The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be consistent with the RRS of specific plant equipment.

g. For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high-frequency excitations. The vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. The use of these prior testing results should be justified by demonstrating that the frequency content of the PSD of the test waveform is sufficient, in accordance with Annex B to IEEE Std. 344-2004.

h. If licensees plan new seismic qualification tests for active mechanical equipment in plants with high-frequency ground motion, the tests should demonstrate the adequacy of the frequency content and the stationarity of the frequency content of the synthesized input waveforms. The frequency content of the Fourier transform of the test waveform or the frequency content of the PSD of the test waveform should be compatible with the amplified portion of the RRS. Annex B to IEEE Std. 344-2004 provides guidelines on frequency content and stationarity.

i. For NPPs that were licensed with the elimination of the OBE, active mechanical equipment qualified by testing should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events, in accordance with Annex D to IEEE Std. 344-2004, when followed by one full SSE (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based on the OBE level, in accordance with the licensing basis.

1.2.2 Specific NRC Staff Positions

The following are specific NRC staff positions, including exceptions and clarifications, on ASME QME-1-2007:

a. Section QR-A6200 (Damping)

The damping values used in analysis should be in accordance with the damping values listed in Table 6 of RG 1.61, Revision 1, or as approved in the plant licensing basis. Damping values other than those provided in the plant licensing or design basis, or in RG 1.61, may be used, subject to NRC review and approval, if documented test data support the higher values.

b. Section QR-A6300 (Seismic Qualification Requirements—Required Response Spectrum)

Section QR-A6300 states: "For in-line active mechanical equipment qualified in accordance with QR-A7400 (Earthquake Experience-Based Qualification), the RRS is typically the building filtered response spectrum at the distribution system support attachments to the building." The use of the building filtered response spectrum at the distribution system support attachments to the building as the RRS for the in-line equipment may not be adequate. The RRS for in-line active mechanical equipment should account for the potential motion amplification of the distribution system.

c. Section QR-A7331 (Qualification by Similarity—Excitation)

Section QR-A7331 states that; "a conservative composite excitation may be generated by extrapolations or interpolations of data whose parameters are not identical but are justifiable." Likewise, excitation whose spectral content are significantly different may be used to generate lower-level composite estimates, providing that an account is taken of possible multi-axis response or cross-axis coupling, or both." The licensee should provide, for NRC review and approval, detailed information justifying this statement.

d. Section QR-A7421 (Earthquake Experience-Based Qualification—Attributes of Equipment Class)

The NRC staff will review, in detail, the attributes of the equipment for establishing the inclusion rules that constitute the reference equipment class for earthquake experience, as described in Section QR-A7421, to determine the acceptability of similarity arguments to define a reference equipment class.

Section QR-A7421 also states the following:

> Prohibited features should include any attributes that would contribute to fatigue failure from low cycle loads. The rules of this section apply to active mechanical equipment that may undergo 5 OBEs or aftershocks and one SSE resulting in 60 full range stress cycles during plant life. If a component contains items which could experience a fatigue failure from low cycle loads (less than 60 full range stress cycles), it shall be evaluated in accordance with Section QR-A6800.

The NRC regulations delineated in 10 CFR Part 100, Appendix S to 10 CFR Part 50, and 10 CFR Part 52 require the demonstration of successful equipment functionality under OBEs.

e. Section QR-A7423 (Earthquake Experience-Based Qualification—Functionality during Earthquake)

Licensees should submit, for NRC review and approval, detailed information about the justification used to demonstrate the functionality of the reference equipment class during and after an earthquake.

f. Section QR-A7431 (Earthquake Experience-Based Qualification—Inherently Rugged Active Mechanical Equipment)

To justify the active mechanical equipment class as an "inherently rugged active mechanical equipment" class, the licensee should provide, for NRC review, information regarding the operational or shipping loads as compared to the expected seismic loads that the equipment could be subjected to, and the explicit design standards applied to this equipment class. Licensees should also provide, for NRC review, detailed information regarding the simplified and reduced rules, including the technical justification and data for characterizing the inherently rugged active mechanical equipment class and the procedure for defining the seismic capacity for this equipment class (i.e., the earthquake experience spectrum).

g. Section QR-A7432 (Earthquake Experience-Based Qualification—Limitations)

The limitations for the use of an earthquake-experience-based method of seismic qualification of equipment, as described in Section QR-A7432, may not be a complete list. The list should be expanded to include additional limitations as a result of new findings from testing new equipment or new studies.

h. Section QR-A7440 (Earthquake Experience-Based Qualification—Qualification of Candidate Equipment)

In-structure response spectra used as the RRS for the qualification of candidate equipment should be in accordance with the licensing or design basis or SRP Section 3.7.2, as applicable. Licensees should submit, for NRC review and approval, any uses of RRS other than those described in the licensing or design basis.

i. Section QR-A8330 (Qualification Report—Earthquake Experience-Based Qualification Documentation)

All ASME Class 1, 2, and 3 active mechanical equipment should comply with the ASME Code Section III requirements. The NRC staff recommends adding the following item to a future revision of ASME QME-1:

> (f) compliance with the ASME Code, Section III, requirements for ASME Class 1, 2, and 3 active mechanical equipment.

j. Attachment C to Nonmandatory Appendix QR-A (Qualification of Pumps and Valves Using Natural Earthquake Experience Data)

Attachment C to Nonmandatory Appendix QR-A is based on the guidelines developed by the SQUG for USI A-46 plants. The provisions in the SQUG guidelines rely heavily on earthquake experience data that the staff considered reasonable to verify the seismic adequacy of existing equipment, and the qualification of new and replacement equipment in older, USI A-46 plants only. The NRC staff has not accepted these SQUG guidelines for the seismic qualification of equipment in plants other than USI A-46 plants. The provisions outlined in Section QR-A7400, including the NRC staff's positions noted in Sections C.1.2.1 and C.1.2.2 of this RG, are acceptable for the seismic qualification of active mechanical components. In addition, contrary to the provisions in Section QR-A7400, the introduction to Attachment C states that the data have not been developed to conclusively demonstrate that pumps and valves function properly during earthquakes. Therefore, Attachment C in its current form does not fully comply with the requirements in 10 CFR Part 100. Attachment C also contains an error in Section C-2. The equipment frequency restriction should be greater than 8 Hz instead of less than 8 Hz.

2. Functional Qualification of Active Mechanical Equipment

2.1 Regulatory Positions on ASME QME-1-2007

2.1.1 General NRC Staff Positions

In general, the NRC staff finds ASME QME-1-2007 acceptable for the functional qualification of (1) active mechanical equipment in new NPPs and (2) new or replacement active mechanical equipment in operating NPPs, subject to the following provisions:

a. Appendices

In endorsing the use of ASME QME-1-2007, the staff acknowledged that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, and QP-E; and Mandatory Appendix QV-I and Nonmandatory Appendix QV-A). The staff position is that, if a licensee commits to the use of nonmandatory appendices in ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, then the criteria and procedures delineated in those nonmandatory appendices become part of the requirements for its qualification program, unless specific deviations are requested and justified.

b. Nonmandatory Appendix QR-B

This appendix recommends a methodology and describes the documentation that should be available in a user's files to demonstrate the qualification of nonmetallic parts, materials, or lubricants. It addresses the steps for the user of the active mechanical equipment to follow to qualify and maintain the qualification of the nonmetallic material that is part of the active mechanical equipment. The NRC staff considers Nonmandatory Appendix QR-B to provide a reasonable approach to the qualification of nonmetallic material in active mechanical equipment.

c. Sections QDR and QP

The NRC staff considers Sections QDR and QP to provide a reasonable approach to the qualification of dynamic restraints and active pump assemblies, respectively. These sections have not changed from those in ASME QME-1-2002, and they still adequately document the state of the art of the nuclear industry in the qualification of dynamic restraints and active pump assemblies.

d. Section QV

The revision to ASME QME-1 reflects valve performance information obtained from nuclear industry programs and the NRC's research since the development of ASME QME-1 in the 1980s. With the active involvement of industry personnel and the NRC staff in the development of ASME QME-1-2007, only a few NRC staff exceptions and clarifications are necessary for Section QV, as described in Section C.2.1.2 below.

2.1.2 Specific NRC Staff Positions

a. The definition of "valve assembly" in Section QV-4000, "Definitions," refers to power-operated valves. The NRC staff considers the power actuators for valve assemblies to include all types of power actuators, such as motor, pneumatic, hydraulic, solenoid, and other drivers.[3]

b. Section QV-6000, "Qualification Specification," states that the owner or owner's designee is responsible for identifying the functional requirements for a valve assembly, and that these requirements shall be provided in a qualification specification prepared in accordance with Mandatory Appendix QV-I. The NRC staff considers Mandatory Appendix QV-I to be a necessary part of the implementation of Section QV of ASME QME-1-2007. For example, Mandatory Appendix QV-I provides the definitions of QV Category A and B valve assemblies used in Section QV of ASME QME-1-2007.

3 The guidance in ASME QME-1-2007 may also be used, where applicable, in the qualification of manually operated valves.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC's plans for using this regulatory guide. The NRC does not intend or approve any imposition or backfit in connection with its issuance.

In some cases, applicants or licensees may propose or use a previously established acceptable alternative method for complying with specified portions of the NRC's regulations. Otherwise, the methods described in this guide will be used in evaluating compliance with the applicable regulations for license applications, license amendment applications, and amendment requests.

REFERENCES[4]

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.

3. 10 CFR Part 100, "Reactor Site Criteria," U.S. Nuclear Regulatory Commission, Washington, DC.

4. Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, June 1988.

5. IEEE Std. 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc., New York, NY, January 1987.[5]

6. Regulatory Guide 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Washington, DC, March 1981.

7. American National Standards Institute (ANSI) N278.1-1975, "Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard" American Society of Mechanical Engineers, New York, NY.

8. ASME QME-1-1994, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," American Society of Mechanical Engineers, New York, NY, June 1994.

9. ASME QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," American Society of Mechanical Engineers, New York, NY, November 2007.

10. IEEE Std. 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc., New York, NY, June 2005.

4 Publicly available NRC published documents such as Regulations, Regulatory Guides, NUREGs, and Generic Letters listed herein are available electronically through the Electronic Reading room on the NRC's public Web site at: http://www.nrc.gov/reading-rm/doc-collections/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD; the mailing address is USNRC PDR, Washington, DC 20555; telephone 301-415-4737 or (800) 397-4209; fax (301) 415-3548; and e-mail PDR.Resource@nrc.gov.

5 Copies of the non-NRC documents included in these references may be obtained directly from the publishing organization.

11. IEEE Std. 344-1975, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, Inc., New York, NY, 1975.

12. Regulatory Guide 1.100, Revision 1, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, August 1977.

13. Generic Letter 87-02, “Verification of Seismic Adequacy of Mechanical and Electric Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46,” U.S. Nuclear Regulatory Commission, Washington, DC, February 1987.

14. Generic Letter 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” U.S. Nuclear Regulatory Commission, Washington, DC, June 1989.

15. Information Notice 90-40, “Result of NRC-Sponsored Testing of Motor-Operated Valves,” U.S. Nuclear Regulatory Commission, Washington, DC, June 1990.

16 Topical Report TR-103237, Revision 2, “EPRI MOV Performance Prediction Program,” and Addenda 1 and 2, nonproprietary versions, Electric Power Research Institute, Palo Alto, CA, April 1997

17 Safety Evaluation, “Safety Evaluation on EPRI MOV Performance Prediction Methodology,” U.S. Nuclear Regulatory Commission, Washington, DC, March 1996.

18 Safety Evaluation Supplement, “Supplement to Safety Evaluation on EPRI MOV Performance Prediction Methodology,” U.S. Nuclear Regulatory Commission, Washington, DC, February 1997.

19 Safety Evaluation Supplement, “Supplement 2 to Safety Evaluation on EPRI MOV Performance Prediction Methodology,” U.S. Nuclear Regulatory Commission, Washington, DC, April 2001.

20 Safety Evaluation Supplement, “Supplement 3 to Safety Evaluation on EPRI MOV Performance Prediction Methodology,” U.S. Nuclear Regulatory Commission, Washington, DC, September 2002.

21 Safety Evaluation Supplement, “Supplement 4 to Final Safety Evaluation on Addenda 3, 4, 5, 6, and 7 to EPRI Topical Report (TR)-103237, “EPRI MOV Performance Prediction Program, Revision 2,” U.S. Nuclear Regulatory Commission, Washington, DC, February 2009.

22 Information Notice 96-48, “Motor-Operated Valve Performance Issues,” U.S. Nuclear Regulatory Commission, Washington, DC, August 1996.

23 Generic Letter 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,” U.S. Nuclear Regulatory Commission, Washington, DC, September 1996.

24 Topical Report MPR-2524, “Joint Owners’ Group Motor Operated Valve Periodic Verification Program Summary,” Joint Owners Group, February 2004.

25 Final Safety Evaluation, “Final Safety Evaluation on Joint Owners’ Group Program on Motor-Operated Valve Periodic Verification,” U.S. Nuclear Regulatory Commission, Washington, DC, September 2006.

26 NUREG/CR-6478, “Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing,” U.S. Nuclear Regulatory Commission, Washington, DC, July 1997.

27 Technical Update 98-01, “Actuator Output Torque Calculation,” Limitorque Corporation, Lynchburg, VA, May 1998, and Supplement 1, July 1998.

28 Information Notice 96-48, Supplement 1, “Motor-Operated Valve Performance Issues,” U.S. Nuclear Regulatory Commission, Washington, DC, July 24, 1998.

29 NUREG/CR-6620, “Testing of DC-Powered Actuators for Motor-Operated Valves,” U.S. Nuclear Regulatory Commission, Washington, DC, May 1999.

30 Topical Report NEDC-32958, “BWR Owners’ Group DC Motor Performance Methodology—Predicting Capability and Stroke Time in DC Motor-Operated Valves,” Boiling Water Reactor Owners’ Group, March 2000.

31 Regulatory Issue Summary 2001-15, “Performance of DC-Powered Motor-Operated Valve Actuators,” U.S. Nuclear Regulatory Commission, Washington, DC, August 2001.

32 SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” ADAMS ML No: ML003708021, U.S. Nuclear Regulatory Commission, Washington, DC, April 1993.

33 Regulatory Guide 1.61, Revision 1, “Damping Values for Seismic Design of Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, March 2007.

34 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC.

35 ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” American Society of Mechanical Engineers, New York, NY.

Staff Responses to Public Comments on Draft Regulatory Guide DG-1175
(Proposed Revision 3 of Regulatory Guide 1.100)

(Public comments have been edited for clarity)

IEEE Power Engineering Society Nuclear Power Engineering Committee (ML082000783)	Nuclear Utility Group on Equipment Qualification (ML082000784)	Nuclear Energy Institute 1776 I Street NW, Suite 400 Washington, DC 2006-3708 (ML082540405)
Westinghouse Electric Company P. O. Box 355 Pittsburgh, PA 15230-0355 (ML082000782)	Duke Energy Carolinas, LLC EC07H / P.O. Box 1006 Charlotte, NC 28201-1006 (ML082000785)	Dominion One James Rive Plaza Richmond, VA 23219 (ML082200057)
	ASME Nuclear Codes and Standards (ML082660608)	

Comments			NRC Comment Resolution
Originator	DG-1175 Section	Specific Comment	NRC Staff Response
IEEE-1	B.1	In paragraph (5), we disagree with the NRC comment about the seismic vulnerability of solid state components. Test results experienced by members have shown high capacities for solid state relays. The specific nature of the NRC data showing issues for these types of components should be examined. Recommend deleting these sentences	The staff has reviewed and considered the comment. The statement is revised to "*Some solid-state relays and microprocessor-based components may be sensitive to earthquake excitations.*" A test would be needed to confirm if particular equipment is not sensitive to high-frequency ground motion.
IEEE-2	B.1	In paragraph (5), the term "equipment capacity factor" is not defined. Recommend defining this term.	The staff reviewed the comment and revised the statement in the final version of DG-1175.

IEEE-3	B.1	In paragraph (5), the statement "Third, since no new NPPs were built after the early 1980s, a number of manufacturers for electric or active mechanical equipment are no longer in business, and the appropriateness of using the test experience of old equipment made by manufacturers no longer in business for the seismic qualification of modern equipment designs made by different manufacturers is highly questionable," is misleading and should be corrected or deleted. Many NPPs have been built since the early 1980s. Kashiwazaki-Kariwa Nuclear Power Plants has five BWR units, which entered commercial operation in 1985 and 1994 and two ABWRs which entered commercial operation between 1996 and 1997. There are at least 29 NPPs worldwide that have been built since early 1980s that have utilized IEEE 344 standard for qualification. We also disagree with the suggestion that seismic fragilities are manufacturer-specific. The construction of the equipment and the observed failure modes which are addressed by the similarity requirements in the Standard, are of significantly greater importance. U.S. NRC concern about use of experience data for older equipment is not warranted since such equipment would not be similar to more current components. IEEE Std 344-2004 Section 10.4.2 (a) excludes the use of data for components that have changed in time (such as microprocessor systems). Recommend deleting this entire discussion.	The statement is rewritten as "*Furthermore, since no new NPPs were built in the USA after the early 1980s, a number of manufacturers for electric or active mechanical equipment are no longer in business…"* While the staff agrees that many NPPs have been built outside the USA, not all seismic and test experience database were available to the staff.
IEEE-4	B.1	In paragraph (6) the U.S. NRC concern about using experience data for equipment exposed to harsh environment is not valid. Aging and other environmental requirements are governed by IEEE 323 standard. In addition, EPRI has conducted substantial research (NP3326) to identify those components that do not have a seismic aging correlation.	The staff reviewed the comments and deleted the paragraph in the final version of DG-1175.

		Recommend removing the beginning "The NRC staff - IEEE Std 344-2004.”	
IEEE-5	B.1	In paragraph (7) the statement "Ball joints and kinematics linkages of the shake tables generated these inadvertent high frequencies, which the NRC staff considers to be noise signals that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies" is misleading and should be corrected or deleted. Since the high-frequency range has been characterized as (20 hertz (Hz) and above), RRS used in seismic testing envelope plant equipment location requirements almost always exceed 20 Hz and often contain higher than 33 Hz content purposely input into the seismic test table, there is no basis to state with certainty that "Ball joints and kinematics linkages of the shake tables generated these inadvertent high frequencies". Recommend removing this discussion.	The staff has reviewed and considered the comment. Even though IEEE Std 344-2004 may have safeguards to ensure that the input is generated and in compliance with the frequency range of interest, the statement is needed to prevent potential misuse of the previous test data. The statement is revised to “*Therefore, any attempt to use such past test experience data for the seismic qualification of high-frequency-sensitive equipment or fragile components in such plant is not appropriate unless frequency content of the power spectral density (PSD) of the test waveform has been evaluated in accordance with Annex B of IEEE 344-2004*”.
IEEE-6	B.1	In paragraph (7) the statement "However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz, although the TRS may have shown a zero period acceleration (ZPA) up to 100 Hz" is misleading and should be corrected or deleted. The statement intertwines two seismic qualification elements and generates a misunderstanding. The seismic test frequency range is the amplified range, which is defined by the RRS. The ZPA is by definition the acceleration level of the high-frequency, non-amplified portion of the response spectrum. Recommend removing this discussion.	The staff has revised the statement for clarification as “*although the TRS may have shown a zero period acceleration (non amplified frequency range) up to 100 Hz*”.

IEEE-7	B.1	In paragraph (7) per U.S. NRC concurrence, new plants are not being qualified for high frequency ground-motions rather they are being screened for high frequency sensitivity. Such high-frequency motions are not part of the certified design basis. Refer to COL/DC-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Codification and Combined License Applications" Recommend removing the discussion beginning "When new - motion concerned."	All equipment in new nuclear plants must satisfy the regulations for seismic qualification delineated in Appendix A of 10 CFR Part 100 and Appendix S of 10 CFR Part 50. The staff acknowledges that there is no inconsistency between COL/DC-ISG-1 and DG1175. The ISG provided guidance on the methodology to determine whether the equipment is sensitive to the effects of high frequency ground motion. DG-1175 described methods that the staff considered acceptable for use in seismic qualification of electric and active mechanical equipment. IEEE Std 344 is mentioned in the ISG-1 twice. In Section 4.1.1 of COL/DC-ISG-1, "*If existing test data are used to demonstrate functionality, the use of such data should be evaluated over the required frequency range of interest in accordance with IEEE Standard 344 to demonstrate that the proper frequency content with sufficient amplitude was used as input to the component that has been previously tested*". For the screening procedure and justification of high frequency sensitive equipment, requirements in IEEE Std 344 should be used to demonstrate that the proper frequency range and sufficient amplitude was used. Section 4.3.1 in COL/DC-ISG-1 indicated that "*The test procedure is to be consistent with the requirements of IEEE-344 as supplemented by NRC RG 1.100*". For seismic qualification of screened-in equipment/components, any test procedure should be consistent with IEEE Std 344.

IEEE-8	B.1	In paragraph (7) U.S. NRC defines high-frequency range as 20 Hz and above. It is understandable that a bound was not defined because it is dependent on the frequency range of interest of the hard rock site. Recommend adding a statement in this section to define the upper limit to the high frequency range.	The staff agrees that the bound of the high-frequency range depends on the frequency range of floor response spectrum of the hard rock site. Defining an upper limit would not be appropriate in the guidance document. Thus, no change is necessary in the final version of DG-1175.
IEEE-9	B.1	In paragraph (7) it appears that the NRC position in this section is that previous seismic test programs which did not require HF content cannot be used for qualification of equipment at HF sites. Is it the NRC position that only seismic test programs that required HF content (i.e., hydrodynamic loadings associated with BWR) are acceptable? All seismic tests should be acceptable provided there is sufficient energy content over the frequency range of interest. Recommend this section be revised to be consistent with COL/DC-ISG-1, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications."	See responses to IEEE-5 and IEEE-7
IEEE-10	B.2	In paragraph (1) the major change from ASME QME-1-2002 to ASME QME-1-2007 in terms of the functional qualification of mechanical equipment is a complete rewrite of Section QV and the new Mandatory Appendix QV-1. This entire section seems out of place in a seismic qualification document. This material addresses functional qualification and may be a better fit in Regulatory Guide (RG) 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to safety in nuclear Power plants." RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. Although the ANSI standard by itself does not provided complete assurance of operability, there is an overlap between DG1175 and RG 1.148 for functional qualification of	The NRC staff plans to withdraw Regulatory Guide (RG) 1.148 as soon as this revision to RG 1.100 is finalized. RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. In the Foreword of ASME QME-1-2007, it was explained that the ANSI N45 Committee's valve task force (N278) was reassigned to the ASME QME in 1982 and designated the Subcommittee on Qualification of Valve Assemblies. In addition, ANSI N278.1 has not been updated since 1975 and the staff believes that there is no need to revise RG1.148. Endorsing the ASME QME-1-2007, which incorporated all the lesson-

		active valves. Recommend that functional qualification of active mechanical components (which have no direct bearing on seismic qualification) should be discussed in a revision to RG 1.148 and the RG 1.100 should only provide guidance for seismic qualification of electric and mechanical equipment. Therefore, Section 2. (Functional Qualification of Active Mechanical Equipment) should be move to RG 1.148 and the title for this document reverted back to "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants	learned and operating experience of active mechanical equipment, for functional qualification is appropriate and prudent.
IEEE-11	C.1.1.1b	In subsection (1) of C.1.1.1b the word "credibility" is used. This word may imply a negative bias and is not suggested for use in a regulatory position document. The following change is recommended: 1) The wordings "the credibility and" be removed. The remaining wording is sufficient. or 2) Change to the following: (1) seismic experience data for its completeness and the information that would be generated in the process of establishing evidence of qualification.	The staff disagrees with the comment. Not all test and earthquake experience data have equal technical quality. The credibility, or the quality of the data, should be justified.
IEEE-12	C.1.1.1c	This sub-section states "The NRC staff does not generally find it acceptable to use experience data (earthquake or test experience data) for ..." and goes on to provide three categories of equipment which are very extensive and encompass the majority of safety-related electrical and electromechanical equipment provided to Nuclear Power Plants (NPPs). It is unclear why the NRC find experience based qualification to be an unacceptable method. As written the DG-1175 position suggests that test-based experience performed in accordance with IEEE Std 344-2004 requirements (per Section 10.3) does not adequately qualify chatter sensitive equipment. This is unclear since sub-clause	The staff does not disallow the use of experience-based methods. As delineated in C.1.1.1b, the use of experience-based method for seismic qualification of electric equipment will be subject to the review and approval by the NRC staff. Even though IEEE Std 344-2004 identified limitation of earthquake or test experience – based qualification, the staff believes that the list in IEEE Std 344-2004 sub-clause 10.4.2(b) should be supplemented by the additional items listed in C.1.1.1c. However,

		10.4.2 (b) of IEEE Std 344-2004 provided exclusion to such things as relays, contactors, switches and breakers. Experience based method as defined in IEEE Std 344-2004 provides sufficient evidence of seismic qualification. Further clarification is recommended to understand the DG-1175 position.	C.1.1.1c should not be considered as a complete list.
IEEE-13	C.1.1.1c	Please clarify what are fragile electronic components, such as solid-state relays and microprocessors-based components This paragraph provides an inappropriate conclusion that safety-related solid-state relays and microprocessor-based components are fragile. Test results experienced by IEEE Std 344-2004 Working Group (WG) have shown high capacities for equipment like solid-state relays. Safety-related solid-state relays and microprocessor-based components have been seismically qualified to IEEE Std 344-2004 by testing and have recently experienced actual earthquakes, such as the Niigataken Chuetsu-Oki earthquake at the Kashiwazaki-Kariwa Nuclear Power Plant in which safety-related digital I&C operated properly during and after the earthquake. Seismic qualification test programs and earthquake experience demonstrated that safety-related solid-state relays and microprocessors-based components are not fragile. Therefore, the characterization "fragile electronic components" must mean "non-safety-related" solid-state relays and microprocessor-based components. Recommend DG-1175 data identifying “fragile electronic components" be provided for review and the statement further clarified or deleted since as written it is misleading.	The staff has reviewed and considered the comment. The statement is revised in the final version of DG-1175
IEEE-14	C.1.1.1c	Item (3) identifies a concern with the using of experience data for subcomponents that are defined in Items (1) and (2). This exclusion is presently addressed in IEEE Std 344-2004 in the Introduction and the exclusion defined in sub-clause 10.4.2 (b). Therefore,	The staff has reviewed and considered the comment. The statement is revised in the final version of DG-1175

		since this item is addressed in. IEEE Std 344-2004, no restriction is required. Recommend this exclusion be removed since it is presently addressed in IEEE Std 344-2004.	
IEEE-15	C.1.1.1d	This sub-section as written seems to impose new requirements on the common practice of seismic testing selected items to qualify a family of similar items in accordance with Claus 8 (Testing) of IEEE Std 344-2004. If so, this is a change from traditional seismic qualification methods used in the past. Recommended this statement be further clarified to better define the intent of the section and the relationship to similarity method defined in sub-clause 9.3 (Extrapolation for similar equipment) of IEEE Std 344-2004 or be deleted. This section should allow the industry to qualify similar equipment without having to obtain NRC approval.	The SQUG concluded, and the NRC agreed, that the use of experience data was feasible for the purpose of verifying equipment seismic adequacy for the older vintage USI A-46 plants. Thus, the older vintage USI A-46 plants can, with a few exceptions, use the experience-based SQUG-GIP method for seismic verification of new and replacement equipment provided they have revised their licensing bases via safety evaluations. C.1.1.1d is applicable only to applicant or licensee who is proposing to use test-experience data (in IEEE St 344-2004 Clause 10.3) to perform seismic qualification. C.1.1.1d is not applicable to the provision in Clause 8 (Testing) and Clause 9 (Combined analysis and testing).
IEEE-16	C.1.1.1f	Seismic qualification of equipment should be performed over the frequency range of interest. DG-1175 wording does not allow a limit lower than 33 Hz to be performed but mandates a higher cutoff is required by the RRS of a specific plant. There may be instances where a lower cut-off would be allowed by a site specific RRS and therefore should be allowed. IEEE 344 standard uses the following wordings throughout the standard to address this item. "...over the frequency range of interest (typically, 1 Hz to 33 Hz)" or "up to the cutoff frequency." Where the cutoff frequency is defined as "The frequency in the response spectrum where the ZPA asymptote begins..."	The staff agrees the wording in IEEE Std 344-2004 concerning the frequency range has not changed from the IEEE Std 344-1987 version. However, recent studies identified that the response spectrum for certain sites in the Central and Eastern United States may have amplified region in the beyond 33 Hz. The staff agrees that defining an upper limit would not be appropriate in the guidance document. The upper bound of the frequency range should be depends on the frequency range of the RRS of the specific plant equipment. The statement is revised to "*The NRC staff does*

		The IEEE Std 344-2004 wording above is appropriate. The wording has not changed from the IEEE Std 344-1987 version. Recommend removing this discussion.	*not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be consistent with the RRS of specific plant equipment*."
IEEE-17	C1.1.1g	This section excludes the use of previous testing to address high frequency concerns because the high frequency motions were not intentionally input to the test. An assessment of the test input waveform should be conducted to verify the test specimen was adequately tested over the frequency range of interest. If the test data demonstrate sufficient frequency content in the high-frequency range throughout the time history then the data should be acceptable. This approach is consistent with Section C.1.1 1.h. IEEE Std 344-2004 has sufficient safeguards to ensure that the input is generated and in compliance with the frequency range of interest. The DG-1175 position is not necessary since the present requirements in IEEE Std 344-2004 are adequate to verify the test data has sufficient content over the frequency range of interest throughout the input time history. This requirement is addressed in the stationary requirements in the strong motion portion of the test inputs through time segment analysis as defined in IEEE Std 344-2004 Annex B. If there is sufficient content in each of the time segments then the test input is acceptable and the origins of the energy input to the test (ball joints and kinematic linkages) are immaterial. The test specimen experienced the required environment regardless of source. Recommended this section be revised to require high frequency motions evaluated in accordance with IEEE	The staff has reviewed and considered the comment. Even though IEEE Std 344-2004 may have safeguards to ensure that the input is generated and in compliance with the frequency range of interest, the statement is needed to prevent potential misuse of the previous test data. The statement is rewritten as "*The vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. The use of these prior testing results should be justified by demonstrating that the frequency content of the power spectral density (PSD) of the test waveform is sufficient in accordance with Annex B of IEEE 344-2004".*

		Std 344-2004 Annex: B (Frequency Content and Stationarity).	
IEEE-18	C.1.1.1i	The statement "Electric equipment should be qualified with five one-half SSE events followed by one full SSE event (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less," should be deleted since the statement as-is creates a situation where current acceptable testing may be rendered unacceptable. The DG-1175 position does not recognize that some plants are licensed with an OBE that is greater or less than one-half the SSE. The plant licensing basis should define whether the OBE is one-third or one-half of the SSE, or has no relationship to the SSE. The SECY-93-087 document specifically addressed issues affecting Advanced Light-Water Reactors (ALWRS), for which the OBE was eliminated as a design case by making it one-third of SSE or less. The five one-half SSEs provision in SECY-93-087 is intended for ALWR applications. It is also noted that the DG-1175 phrase ",..even if the OBE of a plant is defined to be one-third of SSE or less" is not in SECY-93-087. The OBE tests in IEEE 344 standard are intended to simulate vibratory aging effects for conditions where plant operation is expected to proceed without requiring shutdown. Recommend this section be revised to reflect that the OBE amplitude should be based on the applicable plant licensing requirements.	The statement is revised for clarification. *"For NPPs that were licensed with the elimination of the OBE, electric equipment should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events in accordance with Annex D of IEEE 344-2004 when followed by one full SSE (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based the OBE level in accordance with the licensing basis".*

IEEE-19	C.1.1.1j	This section states "The IEEE Std 344-2004 recommended no damping values." This is not correct as written. IEEE Std 344-2004, Clause 6 (Damping) provides specific details regarding the application of damping. Sub-clause 6.3.1 (Application of damping in analysis) identifies "Appropriate values of damping may be obtained from tests or other Justifiable sources" Further clarifications are provided in subclauses 6.3.2 (Application of damping in testing) and 8.6.1.3 (Damping selection) to provide additional guidance on the damping to be used for testing. It should be noted that U.S. Regulatory Guide 1.61, which provides acceptable damping values for seismic analysis and design, also allows for higher damping values if test data is available to support. Recommended this statement be reworded to say IEEE Std 344-2004 recommends appropriate values of damping for analysis may be obtained from tests or other justifiable sources" or deleted since it incorrectly states that IEEE Std 344-2004 does not recommend damping values."	The statement is revised for clarification. "*The damping values used in analysis should be in accordance with the damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 31) issued in March 2007, or as approved in the plant licensing basis.. Damping values other than those provided in the plant licensing/design basis or RG 1.61 may be used, subjected to staff review and approval, if documented test data supports the higher values."*
IEEE-20	C.1.1.2a	This section addresses the susceptibility of safety-related equipment to low cycle fatigue. Low cycle fatigue is the result of materials experiencing structural damage when subjected to cyclic loading. Low cycle fatigue is related to structural integrity which can indirectly affect functionality. Low cycle fatigue susceptibility is a material property that can be screened out. Functionality is a separate issue and IEEE Std 344-2004 requires a separate evaluation for it. Since earthquakes impose repeated cyclic loadings on SSCs, the possibility of fatigue has been identified as a potential failure mechanism. The potential for such failure mechanisms is relatively small because	The staff disagrees. The section addresses not only the low cycle fatigue but also all the possible failure modes that will affect the functionality of the equipment under OBE excitation. The guidance of using five OBE and one SSE, or the equivalent, has been a consistence staff position for seismic qualification of electric and mechanical components to meet the regulations in Appendix A of 10 CFR Part 100 and Appendix S of 10 CFR Part 50.

		earthquakes create only a few cycles of strong motion and most materials are not susceptible to low-cycle fatigue (typically only brittle materials are susceptible to low-cycle fatigue). IEEE Std 344-2004 recognizes this situation in sub-clause 7.6 (OBE and SSE Analysis) where it limits the scope of what is necessary for analysis to only low-cycle fatigue-sensitive equipment: *"The number of OBEs and the fatigue-inducing potential per OBE is important only for low-cycle fatigue-sensitive equipment."* However, DG-1175 does not take exception to the underlying premise for performing repeated OBE tests or analyses. Instead five OBEs are arbitrarily imposed, even though there are other ways to address this issue. One other method for addressing the potential for low-cycle fatigue is to exclude use of experience data for low-cycle fatigue-sensitive equipment as required in IEEE Std 344-2004 subclauses 10.2.3.1 and 10.3.3.1. Recommend this section be revised to remove discussion on low cycle fatigue.	
IEEE-21	C.1.1.2c	The capacity derived from earthquake experience data is an average capacity from many samples. It is appropriate to compare it to an "average" demand such as median-centered. It would also be overly conservative to require the RRS be developed using normally conservative analytical approaches in RG 1.122 and also implement the conservative assumption of the ground motion for the experience data earthquakes to represent the capacity for tie class. In a manner similar to modern code development there should be relative consistency in margin between all approaches. Therefore, the use of conservatively calculated demand (e.g., RG 1.122) is inappropriate. Recommend this section be deleted.	The staff reviewed the comment and revised the statement to "*In-structure response spectra used as the RRS for the qualification of candidate equipment should be in accordance with the licensing/design basis or the Standard Review Plan, (Ref 33), Section 3.7.2, as applicable. The use of RRS other than those described in the licensing/design basis should be submitted for NRC staff review and approval.*"

IEEE-22	C.1.1.2d	This staff position on the first paragraph of sub-clause 10.3.2 imposes the requirement to divide the Test Experience Spectrum (TES) by a factor of 1.4 and cites References 32 and 33 as the basis, The proposed equipment capacity factor of 1.4 is not applied as a capacity reduction factor. Rather, it is applied as a demand increase factor; i.e., one increases the seismic demand by 1.4 and compares to the capacity determined by test or test experience to demonstrate adequate margin in order to meet a stated performance goal. The IEEE 344 standard is intended for equipment qualification in a deterministic evaluation for meeting a design requirement rather than in an evaluation for meeting a probability-based performance goal. IEEE 344 standard has never specified a numerical value of test margin. Instead it simply states that the qualification specification should state what margin is required and refers to IEEE 323 standard, which currently has the suggested margin of 1.1. The RRS, including any required margin, is part of the qualification specification and any margin is controlled by documents external from IEEE 344 standard. In fact, in IEEE Std 344-2004 sub clause 10.3 (Test experience data) is consistent with clause 8 (testing) in that both require seismic demand (Required Response spectrum) to be based on conservative design response spectra rather than realistic median spectra as for Section 10.2 (earthquake experience data). This difference recognizes the relative levels of confidence for qualification by test or test experience and qualification by earthquake experience that was the intent in References 32 and 33. The staff position relative to the second paragraph of sub-clause 10.3.2 does not recognize this section	The staff reviewed the comments and revised the statements. A factor of 1.4 will not be imposed in the final version of DG-1175. The statement is revised to "*The TES shall be the frequency-by-frequency mean of the response spectra from successful tests without malfunction. When using test experience data, both the mean and the standard deviation of the data leading to the TES curve should be provided for review and approval. .*" The staff did not take any exception in Clause 8 (Testing) for this issue. The specific staff position C.1.1.2.d.ii is necessary because new reactors are licensed with the elimination of OBE when the OBE is 1/3 or less of the SSE, not ½ the SSE as indicated in subclause 10.3.2

		requires the items in the reference equipment class must be tested with five OBE and one SSE, as per current staff guidance. This sub-clause was intended to define the requirement for when the development of an OBE test experience spectra TES is required. Recommend this DG-1175 discussion be deleted and that References 32 and 33 not be cited.	
IEEE-23	C.1.1.2e	This section provides a restriction for test experience data that the tested equipment be so similar to each other (1/6 octave) that it becomes a one to one similarity qualification process. The basis of the requirement of 1/6 octave range for class definition natural frequency is very restrictive and not understood. If the plant's licensing basis (especially older operating plants) would allow data analyzed at 1/3 octaves then such criteria should also be acceptable for test experience data. Recommend deleting this discussion.	The use of 1/3 octave will miss the identification of the natural frequency of the equipment and devices especially in the high-frequency range. Thus, no change is necessary in the final version of DG-1175.
IEEE-24	C.1.1.2g	This section states that median-centered horizontal in-structure response spectrum as the RRS for the candidate equipment is not acceptable. The median-centered RRS are not used with Test Experience Data and not referenced in sub-clause 10.3.4 of IEEE Std 344-2004. Recommend deleting this discussion.	The staff has reviewed the comment and agreed that subclause 10.3.4 does not identify the method of developing the in-structure response spectrum to be used with Test Experience Data. The staff has deleted this guidance in the final version of DG1175.
IEEE-25	C.1.1.2k	1.0 SUMMARY OF ISSUE The NRC in DG-1175 has recommended changes to the Coherence and Correlation limits on shake table testing performance that are contained in IEEE Std 344-1987 and IEEE Std 344-2004 versions. The following sections address our technical issue with the position in DG-1175 and provide the rational for why the values should not be changed. NRC comments on the following points are requested, as well as technical justification for the basis	The staff reviewed the written comments and input from the public meeting. The staff has deleted this guidance.

		under which the NRC would recommend such reductions. 2.0 The current Coherence and Correlation limits have been used for over 20 years. This issue involves the performance of seismic shake tables with multiple degrees-of-freedom (DOF). In a biaxial table, for example the motion may be in the horizontal (X) direction and independently in the vertical (Z) direction. For reasons discussed below it is desired that the motion in these two directions not be too similar. That is they must have limited Correlation or Coherence. TECHNICAL BACKGROUND ON COHERENCE AND CORRELATION The Coherence function is a frequency dependent function describing the similarity of two signals on a frequency by frequency basis. By mathematical definition the Coherence function is real valued between 0.0 and 1.0. If two time histories, X and Y, have a Coherence Function of 1.0 for all frequencies of interest they are essentially identical and are totally "coherent". They are very similar. If they have a Coherence function of 0.0 for all frequencies of interest then they are very different and independent from each other (they are not coherent at all). If, for example, they have Coherence in a particular frequency range around 0.5 then they are somewhat similar to each other in this frequency range (somewhat coherent). For reasons discussed below IEEE-344 standard has held that the perpendicular motions on a shake table should have Coherence equal or less than 0.5 at all frequencies of seismic interest. (This typically means between 1 Hz and 33 Hz.) The Correlation Factor of two signals is related to the Coherence Function mathematically but is a single real valued number between -1.0 and +1.0. Generally the absolute value of the Correlation Factor is used and it	

		runs between 0.0 and 1.0. Again signals with a Correlation factor of 0.0 are very different and those with a Correlation factor of 1.0 are essentially identical. For reasons discussed below IEEE Std 344-2004 holds that the Correlation Factor between perpendicular motions on a shake table must be equal to or less than 0.3. IEEE Std 344-2004 specifies that either of these two above criteria must be met for the shake table test to be valid. That is: either the Coherence must be less than or equal to 0.5 at all frequencies of interest or the Correlation Factor need be less than 0.3. Both criteria need not be passed, just one or the other. This test must be done between all pairs of perpendicular motions on the shake table. For a biaxial table this refers to the X and Z directions. For a triaxial table this refers to the X and Z, X and Y (where Y is the other horizontal direction), and Y and Z directions. 3.0 TECHNICAL BACKGROUND ON EARTHQUAKE SUGGESTED LIMITS These requirements on Coherence and Correlation came from ASME Paper 83,PVP-22, "Suitability of Synthesized Waveforms for Seismic Qualifications" and others in the IEEE 344 WG. The concern leading to this work and suggested limits was that some early shake tables had certain inadequacies that could potentially lead to un-conservative testing. At the most extreme would be the attempt to present a Vector Biaxial table as a true Independent biaxial table. A Vector Biaxial table runs in a single direction (single axis, single DOF) but this axis is tilted with respects to the X and Z axes, for example. Hence such a table can produce both X and Z motion, but these two motions would be nearly identical except for a scaling factor depending on the angle of tilt. In this case the XZ Coherence and Correlation would be nearly 1.0. Hence, the IEEE 344	

		standard limits would clearly invalidate calling such a table an independent triaxial table. (Note that IEEE 344 standard does allow the use of Vector Biaxial tables, but under limited conditions and with test level penalties that do not apply to Independent Biaxial tables.) A second and more subtle concern was that the dynamic stiffness or control system of an independent biaxial (or triaxial) table was insufficient and allowed either table resonances or test item resonance feedback to significantly distort the test motions. This could also occur if the artificial time histories generated to drive the table were inadequately prepared. If this was the case then it was possible that, in a certain frequency range, the X and Y motions could be independent but in the region near the resonances they would be dependent (similar). One can postulate certain cases for certain structures, where such similarity could cause an under test. That is the Test Response Spectra (TRS) measured on the table would, in both directions meet or exceed the Required Response Spectra (RRS), but that certain modes of test object vibration would not be sufficiently excited. Hence, the IEEE-344 WG felt that there must be some limitation on the amount of similarity allowed in a valid test. To answer what kind of limitation the ASME Paper 83-PVP-22 studied the actual correlation between several measured earthquake ground motions. It evaluated the Coherence Function and correlation Factor as these actual earthquakes, and concluded that actual earthquakes do in fact have some non zero values of these factors. The ASME Paper 83-PVP-22 then argued, and the committee and technical community at large eventually accepted, that the restrictions placed on shake tables should be similar to the values found in these actual earthquakes. Review of the actual earthquake data suggested that the value of	

		0.5 for Coherence and 0.3 for Correlation were reasonable. Hence these factors are based on study of actual earthquake ground motion properties. The values of 0.5/0.3 are slightly rounded up averages of the actual earthquake motions in ASME Paper 83-PVP-22. These numbers and concepts are presented in Annex E of IEEE Std 344-2004 and earlier version. The recent NRC recommendations suggest that these limits should be reduced from 0.30 for Correlation Factor. In addition the NRC recommends a Coherence function between 0.0 and 0.3 with an average of 0.2. This reduced from the current IEEE Std 344-2004 requirement of 0.5. 4.0 FIRST REASON FOR NOR REDUCING THESE LIMITS As these limits reflect actual earthquake behavior, reducing the shake table limits further appears unfounded. Reducing these limits would, in a sense, make the shake table tests less "earthquake-like" not more. No reason has been presented why further reduction of the limits is needed, or why this would lead to a more conservative test. It is unlikely that this reduction would provide any significant increase in conservatism or quality of test. 5.0 IN-STRUCTURE SPECTRA - THE SECOND REASON FOR NOT REDUCING LIMITS The ASME Paper 83-PVP.22 study used actual ground motion data. Most equipment is tested to RRS computed in structures. The intervening structure often has resonances that significantly increase the energy content in a selected frequency bands. Further they often do so in all directions. Hence we are often faced with RRS that have strong energy peaks at the same narrow frequency bands in both X and Y directions.	

		These concentrated energy peaks correspond to time signals that are not quite sinusoidal and not fully random. Two sinusoidal or nearly sinusoidal signals at the same or nearly the same frequency are highly correlated. The signals required to match such RRS are, by their fundamental mathematical nature, more highly correlated than the more random ground motions. This phenomenon was recognized in ASME Paper 83-PVP-22. Hence, it becomes difficult, and in some cases, mathematically impossible to simultaneously match such peak RRS and also satisfy low Coherence/Correlation criteria. This is not the result of poor shake table performance. It is the mathematical result of how we process and generate RRS in the nuclear power industry. This issue has plagued IEEE 344 standard shake table testing for years and often made test validation very difficult if not impossible. A further and arbitrary (in our opinion) reduction in Coherence/Correlation limits would only exacerbate this issue to the point of rendering shake table testing impossible. This would be an unfortunate move as in fact the motions found in higher levels of a structure are, in real earthquake, more, not less correlated. Regulation should direct us to use more realistic earthquake motions, not less realistic or mathematically impossible ones. 6.0 ROTATED MOTIONS - THE THIRD REASON FOR NOT REDUCING LIMITS Consider a shake table in which the X and Y perpendicular motions have in fact, somehow, have been generated to have 0.0 Coherence and Correlation. Now consider the motion on this same table at the same time in a set of coordinate's rotated 45 degrees to the X and Y axes. That is, consider two new motions A and B: A = (X + Y) / SQRT(2)	

		B = (X - Y) / SQRT (2) What are the Correlations and Coherence of A and B, which are perpendicular to each other? Assuming X and Y (and hence A and B) are of approximately the same energy level as is typically the case, then the Correlation factor of A and B will be approximately 0.5. Hence when we contemplate trying to reduce the Correlation of shake table motions to near zero in the traditional X and Y axes, we need to remember that even if this task is achievable, the Correlation in a rotated set of axes on the same table will be significantly correlated. Since equipment placement, structural orientation, and direction of earthquakes are somewhat random, there is nothing sacred about the transitional X and Y axes. So in reality, for both real earthquakes and real shake table tests, the motions imparted into the test structure in fact will have and must be somewhat Correlated under some set of axes. This is true even if under a different set of axes the motion is highly uncorrelated. Therefore, we believe it is unreasonable to focus on extreme correlation limits in any one, arbitrary, set of axes. 7.0 PRACTICAL ISSUES - THE FOURTH REASON FOR NOT REDUCING LIMITS Hence real earthquakes are correlated and in-structure earthquakes are even more correlated than we are requiring of our shake tables. Shake table construction and control has been evolving over the years. There are shake tables today that could, with some difficulty, provide lower correlated motions,	

		as long as the limit of mathematical possibility is not crossed. However, there are a number of older shake tables that may not be able to provide lower correlation limits. It would be unfortunate to exclude these tables from performing valuable seismic testing for the nuclear industry based on an unsubstantiated limit reduction. 8.0 ADDITIONAL SUMMARY POINTS AND REFERENCES RELATED TO THESE LIMITS The NRC staff seeks to imposes, in DG-1175, more stringent limits in IEEE Std 344- 2004 Annex E. This suggested limitation and modification of the consensus standard is not consistent with the following sound technical bases identified by the IEEE 344 WG that developed IEEE 344-2004. a. The coherence function and the cross correlation coefficient were originally developed in ASME Paper 83-PVP-22 based on his review of several actual earthquakes. Some of actual earthquakes had factors higher than 0.5/0.3. The recommendation (0.5/0.3) is slightly higher than the average of the actual earthquake results and represents real data. b. The earthquakes in ASME Paper 83-PVP-22 were for free field ground motions. They were not for motions in buildings. ASME Paper 83-PVP-22 noted that ground motions after entering buildings were likely to be more (not less) correlated, due to the multi-directional contribution of many structural modes of vibration. Therefore, it is reasonable to expect that motions on upper floors of a structure will be more, not less, correlated than 0.5/0.3. c. It is unrealistic and nearly impossible to have two real narrow band floor spectra to be less correlated than 0.5/0.3. Requiring motions to have less correlation is unrealistic and mathematically approaching unrealizable	

		d. We have not identified any studies that suggest that a correlation less than 0.5/0.3 results in a significantly more severe test. With current seismic shake tables it will be very difficult, if not impossible, to achieve significantly less than 0.5/0.3. This is caused by a combination of table design/control limitations and the difficulties mathematically in achieving the task. Lowering the 0.5/0.3 criteria would reduce the current seismic test capacity and not achieve any better results. e. The commenter cites regulatory Guide 1.92 revision 1 as providing the NRC staff's position related to the unacceptable nature of using a coherence function of less than 0.5 and cross correlation coefficient of 0.3." Regulatory Guide 1.92, Revision 1 "Combining Modal responses and Spatial Components in Seismic Response Analysis" states In footnote 2 that when using the Time-History Analysis Method, "the earthquake motions specified in the three different directions should be statistically independent. For a discussion of statistical independence, see Reference 6." The reference referred to is a paper in the February 1975 edition of the Journal of the Structural Division, ASCE titled "Definition of Statistically Independent Time Histories." Regulator Guide 1.92 Revision 1 itself does not establish a limiting value for coherence or cross correlation. ASCE standard 4-98 on seismic analysis of safety-related nuclear structures has the following requirement in Section 2.3 on time history input to structures: *“When responses from three components of motion are calculated simultaneously on a time history basis, the input motions in the three orthogonal directions shall be statistically independent and the time histories shall be different. Shifting the starting time of a single time history shall not constitute the establishment of a*	

		different time history. Two time histories shall be considered statistician independent if the absolute value of the correlation coefficients does not exceed 0.3." ASCE standard is an industry consensus standard for seismic analysis of safety- related nuclear structures and is in agreement with the intent of information provided in IEEE Std 344-2004 Annex E.	
IEEE-26	C.1.2.1d	In subsection (1) of C.1.2.1d the word "credibility" is used. This word may imply a negative bias and is not suggested for use in a regulatory position document. The following is recommended: 1) The wording "the credibility and" be removed. The remaining wording is sufficient. Or 2) Change to the following: (1) seismic experience data for its completeness and the information that would be generated in the process of establishing evidence of qualification.	See IEEE-11
IEEE-27	C.1.2.1e	This subsection as written seems to impose new requirements practice of seismic testing selected items to qualify a family of similar items in accordance with QR-A7200 (Qualification by Testing, note that ASME QME has a typographical error and QR-A7200 is mislabeling Qualification by Analysis) of ASME QME-1-2007.If so, this is a change from traditional seismic qualification methods on the common used in the past. It is recommended that the statement be further clarified to better define the intent of this subsection and the relationship to similarity method defined in QR-A7300 (Qualification by Similarity) of ASME QME-1-2007 or be deleted. This section should continue to allow the industry to qualify similar equipment without requiring prior NRC approval.	See IEEE-15

IEEE-28	C.1.2.1f	Seismic qualification of equipment should be deformed over the frequency range of interest. DG-1175 wording does not allow a limit lower than 33 Hz to be performed but mandates a higher cutoff is required by the RRS of a specific plant. There may be instances where a lower cut-off would be allowed by a site specific RRS and therefore should be allowed. That is why the ASME QME standard uses the following wordings "...over the frequency range of interest (typically, 1 Hz to 33 Hz)." The ASME QME-1-2007 wording is appropriate. It is recommended that this statement be reworded or deleted.	See IEEE-16
IEEE-29	C.1.2.1g	This section excludes the use of previous testing to address high frequency concerns because the high frequency motions were not intentionally input to the test. An assessment of the sufficiency of the input waveform should be conducted on the basis of a measurement as defined in ASME QME-1-2007 QR-A7232 or IEEE 344- 2004 Annex B. That will determine whether the component has adequately challenged in all frequency ranges. The origins of the energy input to the test (ball joints and kinematic linkages) are immaterial. DG-1175 does not consider the unintentional vibration due to test table mechanical characteristics to be adequate to meet this requirement even if the ASME QME-1-2007 QR-A7232 or IEEE 344-2004 Annex B frequency content and stationarity requirements are met. The current requirements to demonstrate frequency content and stationarity over the amplified portion of the RRS are adequate, regardless of whether the test table vibrations are intentional, unintentional, or a combination of the two. Recommend this section be revised to require the high frequency motions to be evaluated in accordance with ASME QME-1-2007 QR-A7232 or IEEE Std 344-2004 Annex B.	See IEEE-17.

IEEE-30	C.1.2.1j	The statement "Active mechanical equipment should be qualified with five one-half SSE events followed by one full SSE event (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less," should be deleted since the statement as-is creates a situation where currently acceptable testing may be rendered unacceptable. The DG-1175 position does not recognize that some plants are licensed with an OBE that is greater or less than one-half the SSE. The plant licensing basis should define whether the OBE is one-third or one-half of the SSE, or has no relationship to the SSE. The SECY-93-087 document specifically addressed Light-Water Reactors (ALWRS), for which the OBE was eliminated as a design case by making it one-third of SSE or less. The five one-half SSEs provision in SECY-93087 is intended for ALWR applications. It is also noted that the DG-1175 phrase "...even if the OBE of a plant is defined to be one-third of SSE or less" is not in SECY-93-087. The OBE tests in IEEE 344 standard are intended to simulate vibratory aging effects for conditions where plant operation is expected to proceed without requiring shutdown. Recommend this section be revised to reflect that the OBE amplitude should be based on the applicable plant licensing requirements.	The statement is revised for clarification. *"For NPPs that were licensed with the elimination of the OBE, active mechanical equipment should be qualified with five one-half SSE events followed by one full SSE event or, alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events in accordance with Annex D of IEEE 344-2004 when followed by one full SSE (SECY-93-087) even if the OBE of a plant is defined to be one-third of SSE or less. For other reactors, the staff will review the seismic qualification based on the OBE level in accordance with the licensing basis."*
IEEE-31	C.1.2.2a	The specified damping values in a plant licensing basis	The statement is revised for clarification.

		may be higher or lower than those specified in table QR-A6210-1 or Regulatory Guide 1.61, Revision 1. This subsection should be revised to note that the specified damping values should be in accordance with the plant licensing basis or otherwise determined from testing.	"*The damping values used in analysis should be in accordance with the damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 31) issued in March 2007, or as approved in the plant licensing basis. Damping values other than those provided in the plant licensing/design basis or RG 1.61 may be used, subjected to staff review and approval, if documented test data supports the higher values."*
IEEE-32	C.1.2.2b	Since the Earthquake Experience Spectrum (EES) is based on free field ground motions, and ignores in-structure and in-line amplification at the earthquake site, it is reasonable and conservative to use the demand spectra at the distribution system support location. Further complications of accounting for in-line amplification of the earthquake site facility and the nuclear facility add unnecessary complexity to the qualification. In addition, the nature of the in-line mechanical equipment being discussed is that these equipment classes have adequate variety within the class (supports, frequencies, configurations, etc.) to establish that the in-line amplification is already accounted for within the class. This criterion is adequately addressed in ASME QME-1-2007, and no additional restrictions are required. Therefore, this subsection should be deleted.	The staff disagrees. For piping design, the piping system could be relatively flexible to account for thermal expansion. Due to the flexibility, the amplification of the seismic motion at the equipment location could be very significant depending on the location of the active equipment.

IEEE-33	C.1.2.2d	ASME QME Section QR-A7421 already requires items susceptible to low cycle fatigue failures be evaluated in accordance with QR-A6800, Fatigue and Aging Considerations. The OBE evaluation is performed to consider aging, and it is not required to demonstrate functionality during the OBE. This criterion is adequately addressed in ASME QME-1-2007, and no additional restrictions are required. Therefore, this subsection should be deleted.	See IEEE-20.
IEEE-34	C.1.2.2h	The capacity derived from earthquake experience many samples. It is appropriate to compare it to an "average" demand such as median-centered. It would also be overly consecutive to require the RRS be developed using normally conservative analytical approaches in RG 1.122 and also implement the conservative assumption of the ground motion for the experience data earthquakes to represent the capacity for the class. In a manner similar to modern code development there should be relative consistency in margin between all approaches. Therefore, the use of conservatively calculated demand (e.g., RG.1.122) is inappropriate. This criterion is appropriately addressed in ASME QME-1-2007 and no additional restrictions are required. Therefore, this subsection should be deleted.	See IEEE-21
IEEE-35	C.2	This entire section seems out of place in a seismic qualification document. This material addresses functional qualification and may be a better fit in Regulator Guide (RG) 1.148, "Functional Specification for Active Valve Assemblies in Systems Important to safety in Nuclear Power Plants." RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. Although the ANSI standard by itself does not provided complete	See IEEE-10

		assurance of operability, there is an overlap between DG-1175 and RG 1.148 for functional qualification of active valves. Recommend that functional qualification of active mechanical components (which have no direct bearing on seismic qualification) should be discussed in a revision to RG 1.148 and the RG 1.100 should only provide guidance for seismic qualification of electric and mechanical equipment. Therefore, Section 2. (Functional Qualification of Active Mechanical Equipment) and the title for this document should revert back to "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear power Plants."	
NEI-1	General	The draft RG title and scope have been changed to include functional qualification of active mechanical equipment, as compared to the two previous revisions of RG 1.100 which only discussed seismic qualification of electrical and mechanical equipment. This change is because the RG now endorses ASME QME-1-1994, which covers functional qualification of active mechanical equipment. The main discussion on pages 5 through 8 of the DG is for active, motor-operated valves. It is noted that RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. Although the ANSI standard by itself does not provided complete assurance of operability, there is an overlap between DG-1175 and RG 1.148 for functional qualification of active valves. It is recommended that functional qualification of active mechanical components (which have no direct bearing on seismic qualification) should be discussed in a revision to RG 1.148. RG 1.100 should focus solely on guidance for seismic qualification of electric and mechanical equipment. Remove functional qualification of active mechanical equipment from this DG (address in RG 1.148) such	See IEEE-10

		that RG 1.100 focuses solely on guidance for seismic qualification of electric and mechanical equipment. If this is not done, reconcile the overlap between DG-1175 and RG 1.148 in another manner.	
NEI-2	Page 4, (4th para from top– "Large…") C.1.1.1 b C.1.1.2b,c	In the SERs that NRC sent to the USI A-46 plants in the past, it was stated that older vintage plants could use the experience-based SQUG-GIP method for seismic verification of new and replacement equipment provided they revised their licensing bases. Many older plants are currently using the SQUG-GIP method. The DG is silent on this. Add a sentence at the end of this paragraph to this effect: "However, older vintage plants can, with a few exceptions, use the experience-based SQUG-GIP method for seismic verification of new and replacement equipment provided they revise their licensing bases via safety evaluations." Alternatively, reconcile the fact in the DG that NRC has previously accepted earthquake experience-based qualification of new/replacement equipment in older plants.	The staff has reviewed the comment. The statement in B.1 indicated that the use of experience data was feasible for the purpose of verifying equipment seismic adequacy for the older vintage USI A-46 plants. For clarification, the staff added "*The staff does not accept the use of SQUG guidelines for seismic qualification of equipment in non USI A-46 plants licensed under 10 CFR Part 50 or in plants licensed under 10 CFR Part 52*."
NEI-3	B.1	The middle of the 5th paragraph in Section B.1 says "Some solid-state relays and microprocessor-based components are quite fragile in terms of withstanding earthquake excitations." This is specifically counter to testing experience and counter to the experience of the April 8 IEEE SC2 meeting attendees who were not aware of any experience showing solid-state relays and microprocessor-based components to be particularly vulnerable to earthquake motions. This sentence should be deleted as well as the following sentence based on this conclusion.	See IEEE-13

NEI-4	B.1	The end of the 5th paragraph in Section B.1 says "Third, since no new NPPs were built after the early 1980s, a number of manufacturers for electric or active mechanical equipment are no longer in business, and the appropriateness of using the test experience of old equipment made by manufacturers no longer in business for the seismic qualification of modern equipment designs made by different manufacturers is highly questionable." This specific concern is addressed in IEEE 344 Section 10.3.4h and ASME QR-A7432(a); therefore, this concern is not valid for items qualified in accordance with the two standards. This sentence should be deleted.	See IEEE-3.
NEI-5	B.1 C.1.1.1g C.1.2.1g	The high frequency content, which exists in most existing tests, whether inadvertent or deliberate, will still be imparted to an item on equipment on the shake table. Therefore, high frequency vibratory motions generated on a shake table in an inadvertent manner can be of significance. The DG should clarify that such inadvertent motions can be credited provided they are shown to meet stationarity requirements per Appendix B of IEEE Std 344-1987 or 2004 (when one of these versions of the IEEE Standard is the plant's commitment). However, in IEEE Std 344-1975, there was no requirement for stationarity check. For example, previous seismic shake tests for BWR Mark II and III plants (committed to the 1975 version of the standard) were frequently utilized to qualify equipment for the combined seismic and hydrodynamic loads with high frequency content up to 100 Hz and were accepted by the NRC staff in SQRT audits. Revise to require the high frequency motions to be evaluated in accordance with QR-A7232 or IEEE 344 Annex B, Frequency Content and Stationarity.	See IEEE-5 and IEEE -17

NEI-6	B.1 C.1.1.1i C.1.2.1j	In the last sentence of this paragraph, it says that the test sample shall be subjected to simulated OBE and SSE vibrations. per IEEE Std. 344-2004. In section C.1.1.1i (p. 10) two alternatives for the number of tests/cyclic considerations are provided. However, another alternative when OBE is defined as 1/3 or less of SSE is to use two SSE events (with 10 maximum stress cycles per event) in accordance with SRP 3.7.3 (p. 4), March 2007. The SRP considers this alternative to be equivalent to the cyclic load basis of one SSE and five OBEs. This alternative can save testing duration and should also be listed. Revise these sections to include an option that 2 SSE tests, as an alternative to 5 OBE and 1 SSE are also acceptable when the OBE is designated as 1/3 or less of the SSE.	See IEEE-18 and IEEE-30
NEI-7	C.1.1.1c	This paragraph repeats the inappropriate conclusion that solid-state relays and microprocessor-based components are fragile and suggests that test-based experience performed in accordance with IEEE 344 requirements (per Section 10.3) does not adequately qualify chatter sensitive equipment. Both of these comments are incorrect. These sentences should be deleted	See IEEE-13
NEI-8	C.1.1.1d	This paragraph as written seems to impose new requirements on the common practice of testing selected items to qualify a family of similar items in accordance with IEEE 344 Section 8. This section should be deleted or rewritten.	See IEEE-15

NEI-9	C.1.1.1f, C.1.2.1f	This section states: “The NRC staff does not generally find it acceptable to restrict the frequency range of testing up to 33 Hz. The frequency range should be continued beyond 33 Hz, in accordance with the RRS of a specific plant.” This last sentence could be reworded to provide more clarity. Reword second sentence to read as follows: “For RRS with ZPA frequency in excess of 33 Hz, the frequency range of testing should be accordingly extended to match the RRS.”	See IEEE-16
NEI-10	C.1.1.1i	This section requires that the OBE amplitude be set to 1/2 the SSE, even if the plant license OBE is 1/3 of the SSE. The OBE qualification level should be based on the plant license.	See IEEE-18
NEI-11	C.1.1.1j	The IEEE Std. 344-2004 has a section on damping. While the damping values in RG 1.61 can be used when qualification is by analysis, there should be no specific requirement on damping values to be used for shake-testing, only that the equipment damping at which the RRS is developed should be the same or lower than the TRS damping value. This is not mentioned. Clarify the statement in this section that for qualification by shake-table testing, RRS with any reasonable damping value (such as 5% of critical damping) can be used provided that the TRS is also plotted at the same damping value or a higher damping value.	See IEEE-19
NEI-12	C.1.1.2a	IEEE 344 Sections 10.2.3.1 and 10.3.3.1 provide specific criteria for addressing low-cycle loads. Five OBE tests, or actual earthquakes at the same site, are not the only permitted methods to evaluating low-cycle loads. The standard as written properly imposes those requirements on the qualification.	See IEEE-20

		The section should be deleted.	
NEI-13	C.1.1.2c	The capacity spectra are based on a weighted average of the ground motions, neglecting the in-structure amplification from the experience sites. Therefore, the use of median centered demand spectra results in a conservative capacity/demand comparison. The section should be deleted.	See IEEE-21
NEI-14	C.1.1.2d	Application of the concepts in References 32 and 33 would dramatically revise current qualification practices. For example, the 1.4 factor would have to be applied to every test qualification performed in accordance with IEEE 344 Section 8. The mixing and mismatching of these criteria between the goals of IEEE 344 and References 32 and 33 would need careful consideration and would need to be consistently applied throughout the qualification standard. The criteria in References 32 and 33 need to be deleted from this Section or applied consistently throughout IEEE 344. Without substantial further study, it is recommended that the concepts in references 32 and 33 not be incorporated.	See IEEE-22
NEI-15	C.1.1.2g	This section says that you can not use median centered demand spectra for comparison with the TES. IEEE 344 10.3.4b already requires the use of computed in-structure spectra for the demand as opposed to 10.2.4b which specifies median-centered spectra for comparison with the EES). The section should be deleted.	See IEEE-24

NEI-16	C.1.1.2k	This section requires changing the coherence criteria to lower values. This was discussed in the IEEE 344 Working Group and rejected on sound technical bases as follows: The Working Group believes the criteria established in Annex E are acceptable. Our reasons for objecting to the suggested change are noted below: 1. The coherence function and cross correlation coefficient were originally developed by Kana based on his review of several actual earthquakes. Some of the actual earthquakes had factors higher than 0.5/0.3. The recommendation (0.5/0.3) is slightly higher than the average of the actual earthquake results and represents real data. 2. The earthquakes that Kana used were for free-field ground motions. They were not for motions in buildings. Kana noted that ground motions after entering buildings were likely to be more (not less) correlated, due to the multi-directional contribution of many structural modes of vibration. Therefore, it is reasonable to expect that motions on upper floors of a structure will be more, not less, correlated than 0.5/0.3. 3. It is unrealistic and nearly impossible to have two real narrow band floor spectra to be less correlated than 0.5/0.3. Requiring motions to have less correlation is unrealistic and mathematically approaching unrealizable. 4. We have not identified any studies that suggest that a correlation less than 0.5/0.3 results in a significantly more severe test. With current seismic shake tables it will be very difficult, if not impossible, to achieve significantly less than 0.5/0.3. This is caused by a combination of table design/control limitations and the difficulties mathematically in achieving the task.	See IEEE-25

		Lowering the 0.5/0.3 criteria would reduce the current seismic test capacity and not achieve any better results. 5. The commenter cites Regulatory Guide 1.92 Revision 1 as providing the NRC staff's position related to the unacceptable nature of using a "coherence function of less than 0.5 and cross correlation coefficient of 0.3." Regulatory Guide 1.92, Revision 1 "Combining Modal Responses and Spatial Components in Seismic Response Analysis" states in footnote 2 that when using the Time-History Analysis Method, "the earthquake motions specified in the three different directions should be statistically independent." For a discussion of statistical independence, see Reference 6. The reference referred to is a paper in the February 1975 edition of the Journal of the Structural Division, ASCE, titled "Definition of Statistically Independent Time Histories." Regulatory Guide 1.92 Revision 1 itself does not establish a limiting value for coherence or cross correlation. ASCE standard 4-98 on seismic analysis of safety-related nuclear structures has the following requirement in Section 2.3 on time history input to structures: "*When responses from three components of motion are calculated simultaneously on a time history basis, the input motions in the three orthogonal directions shall be statistically independent and the time histories shall be different. Shifting the starting time of a single time history shall not constitute the establishment of a different time history. Two time histories shall be considered statistically independent if the absolute value of the correlation coefficient does not exceed 0.3*." The ASCE standard is an industry consensus standard for seismic analysis of safety-related nuclear structures and is in agreement with the intent of information provided in IEEE 344 Annex E. 6. The commenter goes on to state that the NRC staff's	

		position on the numerical values for the cross correlation coefficient and the coherence function for defining statistically independent motions are also reflected in Section N-1213.1 of Appendix N of the ASME Section III Code. N-1213.1 states that: "The peak acceleration of the three orthogonal synthetic time histories generally need not occur at the same time. In order to simulate natural earthquake occurrences, the correlation of the synthesized time histories may be evaluated by calculating the cross correlation coefficients and the coherence functions. The artificially generated time histories are acceptable if both their cross correlation coefficients and their coherence functions are approximately equal to the respective functions for past earthquake records. An absolute value of the correlation coefficient less than 0.16 is acceptable. For the coherence function the numerical values ranging between 0.0 and 0.3 with an average of approximately 0.2 are acceptable." Note that this section of the appendix does not prohibit use of coefficients higher than 0.16 or 0.3 and focuses on the goal to have synthetic time histories that are representative of past earthquakes. The current version of IEEE 344 (to which the NRC did not object in Regulatory Guide 1.100) was based on the study of actual earthquakes.	
NEI-17	C.1.2.1d	This section discusses "similarity" between the excitation documented in the experience database and the required seismic excitation. The term "similarity" is too strong as the only spectrum comparison requirement should be that the RRS be enveloped by the test spectrum used in the experience database. Suggest deleting the last part of the last sentence that starts with "as well as similarity between…." Add a sentence to read as follows: "Additionally, the test response spectrum documented in the experience database shall exceed the RRS."	The staff reviewed the comments and acknowledged IEEE 344-2004 provided guidance on this issue. The discussion has been deleted in the final version of DG1175.

NEI-18	C.1.2.1e	This paragraph as written seems to impose new requirements on the common practice of testing selected items to qualify a family of similar items (e.g. valve actuators) in accordance with ASME QME QR-A7200. This section should deleted or rewritten.	See IEEE-15
NEI-19	C.1.2.1j	This section requires that the OBE amplitude be set to 1/2 the SSE, even if the plant license OBE is 1/3 of the SSE. The OBE qualification level should be based on the plant license.	See IEEE-18
NEI-20	C.1.2.2b	Since the EES is based on free field ground motions, and ignores in-structure and in-line amplification at the earthquake site, it is reasonable and conservative to use the demand spectra at the distribution system support location. Further complications of accounting for in-line amplification of the earthquake site facility and the nuclear facility add unnecessary complexity to the qualification. This section should be deleted.	See IEEE-32
NEI-21	C.1.2.2d	QME Section QR-A7421 already requires items susceptible to low cycle fatigue failures be evaluated in accordance with QR-A6800, Fatigue and Aging Considerations. This section should be deleted.	See IEEE-33
NEI-22	C.1.2.2h	The capacity spectra are based on a weighted average of the ground motions, neglecting the in-structure amplification from the experience sites. Therefore, the use of median centered demand spectra results in a conservative capacity/demand. This section should be deleted.	See IEEE-21

NUGEQ-1	B.1, C.1.2.1i	<u>Do Not Impose OBE/SSE Testing for Equipment Also Exposed to Harsh Environments</u> *B.1. Seismic Qualification of Electric and Active Mechanical Equipment (page 5):* *"The NRC staff has a concern regarding electric and active mechanical equipment exposed to harsh environments, aging, and earthquakes. In such cases, the NRC staff does not find it acceptable to use experience data (earthquake or test experience data) for seismic qualification of equipment. The test sample shall be subjected to simulated operating-basis earthquake (OBE) and SSE seismic vibrations in accordance with IEEE Std 344-2004."* *1.2.1 General NRC Staff Positions – i (page 13):* *"For active mechanical equipment exposed to harsh environments, aging, and earthquakes, the staff does not find it acceptable to use experience data (earthquake or test experience data) for seismic qualification of equipment. The test sample shall be subjected to simulated OBE and SSE seismic vibrations in accordance with IEEE Std 344-2004."* <u>NUGEQ Comment:</u> The NRC fails to articulate the technical basis for its "concern" regarding the use of seismic experience data for equipment also exposed to harsh environments and aging. Importantly, the design basis for US plants does not postulate concurrent or sequential seismic and LOCA or HELB events. Consequently, the design basis of these plants does not require that equipment function after an SSE and then a LOCA (or visa versa). The NRC concurred with this fact in prior Regulatory Guide 1.89 comment resolutions but indicated a preference for using the same test sample for both seismic and environmental qualification as a conservative practice. The IEEE acknowledged this fact and reaffirmed the NRC perspectives in IEEE 323-2003 which states:	For section B.1 paragraph 6, the statement "*The NRC staff has two other concerns as well…… in accordance with IEEE Std 344-2004"* has been deleted in the final version of DG-1175. For C.1.2.1i, The staff reviewed the comments and agreed that the use of experience-based methods for equipment exposed to harsh environment, or aging are limited as indicated in Limitations of IEEE Std 344-2004 Clause 10.4.2 (f) and ASME QME-1-2007 Section QR-A7432 (e). This guidance has been deleted in the final version of DG-1175.

		"NOTE—A seismic event is not assumed to occur in conjunction with a loss-of-coolant accident. Rather, the sequence described previously has been developed as the basis of a conservative qualification, not one indicative of a sequence of expected plant events." (IEEE 323-2003 page 10) DG-1175 takes a stated NRC preference for electrical equipment qualification and transforms it into an expectation for mechanical equipment without providing any supportable technical basis. The NRC should provide a coherent basis for its "concern" that warrants establishing this "required" regulatory position. The NUGEQ notes that not all mechanical equipment will be qualified using either experience or OBE/SSE testing. A significant amount of mechanical equipment will be seismically qualified using stress analysis combined with limited but supporting stress tests. The DG-1175 position is silent on the use of such analysis but implies that such analysis is not acceptable since it would direct qualification based on subjecting a test sample to simulated OBE and SSE seismic vibrations in accordance with IEEE 344-2004. Finally, the staff is unclear regarding the significance of "aging" to this position. Virtually all installed active equipment experience some form of in-service aging. Only significant aging mechanisms need to be considered as part of qualification. If the aging is not significant does the stated position permit the use of experience data for equipment whose design basis includes seismic events and harsh environment accidents? NUGEQ Recommendation: Delete the Background and Regulatory Position text which dictates the use of seismic testing to establish seismic qualification for all active mechanical equipment exposed to harsh	

		environments, aging, and earthquakes. Alternatively, the NRC may indicate its preference for the use of OBE and SSE testing for this equipment in lieu of experience data. If the NRC states such a preference then it should also make clear that analysis remains a valid method to seismically qualify such equipment.	
NUGEQ-2	B	<u>Regulatory Guide 1.100 Scope - Limit to Seismic/Dynamic Qualification</u> *B Discussion Background (page 1):* *"The NRC developed this regulatory guide (i.e., Revision 3) to endorse, with exceptions and clarifications, the IEEE Std 344-2004 and the ASME QME-1-2007. (This is the first time the NRC is endorsing ASME QME-1). . . . Sections B.2 and C.2 of this regulatory guide endorse, with exceptions and clarifications, Section QR and the remaining sections of ASME QME-1-2007 (except Nonmandatory Appendix QR-A) for the functional qualification of active mechanical equipment."* <u>NUGEQ Comment:</u> The scope of this proposed revision to Regulatory Guide 1.100 should be consistent with prior versions and should be limited to seismic qualification of mechanical and electrical equipment. The functional qualification provisions of QME-1 should be addressed in separate regulatory guidance, either the Standard Review Plan or a separate regulatory guide, or both. A revision to SRP 3.9.6 "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" and Regulatory Guide 1.48 "Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants" may be the most appropriate methods of addressing the functional qualification provisions of QME-1.	The NRC staff plans to withdraw Regulatory Guide (RG) 1.148 as soon as this revision to RG 1.100 is finalized. RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. In the Foreword of ASME QME-1-2007, it was explained that the ANSI N45 Committee's valve task force (N278) was reassigned to the ASME QME in 1982 and designated the Subcommittee on Qualification of Valve Assemblies. In addition, ANSI N278.1 has not been updated since 1975 and the staff believes that there is no need to revise RG1.148. Endorsing the ASME QME-1-2007, which incorporated all the lesson-learned and operating experience of active mechanical equipment, for functional qualification is appropriate and prudent.

		<u>NUGEQ Recommendation:</u> Limit the scope of RG 1.100 to IEEE 344-2004 and the seismic and dynamic provisions of QME-1 and delete DG-1175 Sections B.2 and C.2. Issue guidance on the functional qualification of active mechanical equipment in separate guidance documents, possibly in SRP 3.9.6 and Regulatory Guide 1.48.	
NUGEQ-3	General	<u>Regulatory Analysis Fails to Evaluate Differences in DG-1175 and SRP 3.9.6</u> *Regulatory Analysis 3. Alternatives Approaches: (page 19)* *"The NRC staff considered the following alternative approaches:* *· Do not revise Regulatory Guide 1.100.* *· Update Regulatory Guide 1.100."* <u>NUGEQ Comment:</u> The NRC has failed to consider the significant differences between the functional qualification provisions of QME-1 as modified by DG-1175 and the recently issued NRC guidance in the March 2007 revision of SRP 3.9.6 "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." These differences suggest significant additional licensee burdens regarding the methods and procedures used to establish functional qualification and the documents used to demonstrate such functional qualification. The March 2007 revision of SRP 3.9.6, without reference to any QME-1 functional qualification provisions and guidance, states: "Conformance with the specific guidance in Subsection II of this SRP section will provide reasonable assurance that the functional design and qualification of pumps, valves, and dynamic restraints within the scope of this SRP section and their associated IST programs satisfy the applicable requirements of 10 CFR 50.55a, particularly the IST program requirements of the ASME Code for Operation	Contrary to the NUGEQ comment, there are no significant differences between the functional qualification provisions of ASME Standard QME-1-2007 and the March 2007 revision of Standard Review Plan (SRP) Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." Both QME-1 and SRP Section 3.9.6 were revised in response to lessons learned from valve performance experience at current operating nuclear power plants, and through NRC and industry research programs. Section B.2 in Draft Guide DG-1175 discusses the valve performance experience that resulted in the preparation of QME-1-2007 and the revision to SRP Section 3.9.6. SRP Section 3.9.6 Acceptance Criterion II.1.B on page 3.9.6-8 states that functional design and qualification of each safety-related pump and valve should be accomplished such that each pump and valve is capable of performing its intended function for a full range of system differential pressure and flow, ambient temperatures, and available voltage (as applicable) under all conditions ranging from normal conditions to design-basis accident conditions. The SRP Section 3.9.6 acceptance criteria are consistent with the provisions in QME-1-2007 to demonstrate that pumps and valves are capable of performing

		and Maintenance of Nuclear Power Plants (OM Code); General Design Criteria (GDC) 1, 2, 4, 14, 15, 37, 40, 43, 46, and 54 in Appendix A to 10 CFR Part 50; Appendix B to 10 CFR Part 50; 10 CFR 52.47(b)(1) and 10 CFR 52.80(a).” The NRC Regulatory Analysis does not evaluate or justify the apparent significant differences and licensee burdens when QME-1 is used in lieu of the existing SRP guidance on functional qualification of mechanical equipment. The NRC determined last year that this SRP guidance meets all applicable regulatory requirements. This burden is exacerbated by the DG-1175 provision, without further analysis or justification, requiring compliance with all the nonmandatory sections of QME-1. <u>NUGEQ Recommendation</u>: The NRC should provide a detailed evaluation and justification for using the more prescriptive provisions of QME-1 in lieu of the existing mechanical equipment functional qualification guidance in SRP 3.9.6 and its referenced documents/standards. This evaluation should include the technical basis for requiring compliance with each of the nonmandatory sections of QME-1. Alternatively, and as suggested in Comment 2 the NRC should limit this revision of Regulatory Guide 1.100 to seismic and dynamic qualification and issued separate guidance on functional qualification of active mechanical equipment.	their design-basis functions. The ASME Standard QME-1-2007 represents one acceptable method to satisfy the acceptance criteria in SRP Section 3.9.6 for the functional design and qualification of pumps, valves, and dynamic restraints, consistent with DG-1175. The NRC staff will evaluate Design Certification applications and COL applications based on the SRP Section 3.9.6 acceptance criteria for the functional design and qualification of pumps, valves, and dynamic restraints. The ASME Standard QME-1-2007, as addressed in DG-1175, provides an efficient and effective approach for satisfying the SRP Section 3.9.6 acceptance criteria. Compliance to the provisions and guidance is optional. The NRC staff will also consider other approaches for the functional design and qualification of pumps, valves, and dynamic restraints proposed by Design Certification and COL Applicants in meeting the SRP Section 3.9.6 acceptance criteria.
NUGEQ-4	C2.1.1a	<u>NRC Should Not Dictate Compliance with Nonmandatory Appendices</u> *2.1.1 General NRC Staff Positions - a: (page 13):* *“In endorsing the use of ASME QME-1-2007, the staff noticed that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-*	The staff agreed that Mandatory Appendices contained provisions must be followed. Compliance of Nonmandatory Appendices, which provided information or guidance, is voluntary. The statement is revised to clarify. *“The staff position is that, if a licensee commits to the use of non-mandatory appendices in ASME*

		C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, and QP-E; and Mandatory Appendix QV-1). The staff position is that, once the user commits to the use of ASME QME-1- 2007 for its qualification of active mechanical equipment in NPPs, the criteria and procedures delineated in those appendices then become the requirements for its qualification program, unless the deviations are justified." <u>NUGEQ Comment:</u> The NUGEQ disagrees with requiring the use of the nonmandatory appendices and believes this may be counterproductive and limit licensee commitments to the use of ASME QME-1. QME-1 makes clear that mandatory appendices contain provisions that must be followed and nonmandatory appendices provide information or guidance that is not imposed. The QME-1 committee has issued several revisions to QME-1 and has clearly determined that the nonmandatory appendices provide information/guidance and do not constitute required elements of the standard. Industry experience with interpreting and implementing QME-1 is needed to refine both the mandatory and nonmandatory portions of QME-1. This lack of experience and recognition that acceptable alternative methods may be available were likely considerations that prompted the QME-1 committee to specify certain appendices as nonmandatory. The NRC states that this is the first time that the NRC is endorsing QME-1. The NRC and many in the industry have little experience interpreting or implementing the provisions of QME-1. This is exemplified by the limited number of QME-1 code cases attached to the 2007 revision. The NUGEQ is concerned that unilaterally dictating implementation of all the nonmandatory appendices represents a significant departure from current accepted	*QME-1-2007 for its qualification of active mechanical equipment in NPPs, then the criteria and procedures delineated in those non-mandatory appendices become part of the requirements for its qualification program, unless specific deviations are requested and justified."*

		industry practices that have been endorsed by the NRC. See for example our subsequent comment regarding Nonmandatory Appendix QR-B, “Guide for Qualification of Nonmetallic Parts.” If the NRC believes it has sufficient experience interpreting the nonmandatory appendices then it may be appropriate for the staff to indicate that the NRC has determined that these appendices represent acceptable methods of complying with QME-1. The NRC needs to be clear that they remain guidance and that other methods may be approved on a case-by-case basis. <u>NUGEQ Recommendation: </u>Delete those portions of DG-1175 that dictate compliance with the nonmandatory portions of QME-1 for licensees that commit to the use of ASME QME-1- 2007. If the NRC accepts the guidance in these appendices then the DG-1175 – NUGEQ Comments 5 NRC should delete the existing language beginning with “The staff position is that, once the user commits to the use of ASME QME-1- 2007” and replace it with the following: “The staff has determined that the contents of these nonmandatory appendices are acceptable for meeting applicable QME-1 provisions for the qualification of active mechanical equipment. Other appropriately justified methods not addressed in these QME-1 appendices may also be accepted on a case-by-case basis.”	
NUGEQ-5	C.2.1.1a	<u>Nonmandatory Appendix QR-B Not Appropriate for All Equipment</u> *2.1.1 General NRC Staff Positions - a: (page 13):* *“In endorsing the use of ASME QME-1-2007, the staff noticed that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-*	The staff agreed that Mandatory Appendices contained provisions that must be followed. Compliance of Nonmandatory Appendices, which provided information or guidance, is voluntary. However, if a user commits to use QR-B for its qualification of active mechanical equipment in NPPs, all the criteria and

		C; Nonmandatory Appendices QP-A, QP-B, QP-C, QP-D, and QP-E; and Mandatory Appendix QV-1). The staff position is that, once the user commits to the use of ASME QME- 1- 2007 for its qualification of active mechanical equipment in NPPs, the criteria and procedures delineated in those appendices then become the requirements for its qualification program, unless the deviations are justified." NUGEQ Comment: The NUGEQ is concerned that requiring compliance with Nonmandatory Appendix QR-B, "Guide for Qualification of Nonmetallic Parts" for all active mechanical equipment will result in excessive and unnecessary procedures, methods, and documentation burdens on licensees for some equipment, including all such equipment located in mild environments. Rigid application of the appendix to all equipment regardless of its plant location or potential for exposure to harsh environmental conditions is inconsistent with existing regulatory guidance. The most recent regulatory guidance regarding environmental qualification of such nonmetallic parts is contained in the March 2007 revision of SRP 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." SRP 3.11 states in part (page 3.11-2): "For mechanical equipment located in a harsh environment, compliance with the environmental design provisions of GDC 4 are generally achieved by demonstrating that the non-metallic parts/components are suitable for the postulated design basis environmental conditions." "For electrical and mechanical devices located in mild environments, compliance with the environmental design provisions of GDC 4 are generally achieved and demonstrated by proper incorporation of all relevant environmental conditions into the design process,	procedures that delineated in both the Mandatory Appendices and QR-B then become the requirements for its qualification program. Justification must be provided for any deviations, which will be subjected to NRC staff review and approval.

		including the equipment specification." SRP 3.11 also states that while environmental design requirements apply to all equipment important to safety (i.e., both mild and harsh environments) that environmental qualification is verification of design, limited to demonstrating that DG-1175 – NUGEQ Comments 6 electrical or mechanical or I&C equipment are capable of performing their safety function under significant environmental stresses (i.e., harsh environments) resulting from design basis events in order to avoid common-cause failure. Regarding mechanical equipment SRP 3.11 makes important distinctions between the methodologies and documentation expectations for harsh and mild mechanical equipment. In both cases the SRP 3.11 establishes flexible guidance and does not dictate the more restrictive methods and documentation provisions contained in Appendix QR-B. SRP 3.11 indicates that for mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) and verifies that the licensee has identified the equipment's location, service parameters, and nonmetallic material capabilities, and has evaluated the environmental effects. For mechanical equipment located in mild environments SRP 3.11 indicates that acceptable environmental design can be demonstrated by the "design/purchase" specifications containing a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. In contrast, it appears that Appendix QR-B would dictate that the equipment qualification report for all affected equipment regardless of location (i.e., harsh or mild) contain detailed information on the equipment's	

		nonmetallics, including their function, formulation identification, activation energy, service conditions, failure modes and aging significance evaluations, qualification basis, qualified life, and maintenance/replacement requirements. While such information is similar to that developed to achieve compliance with 10 CFR 50.49 for electrical equipment located in a harsh environment, it is not required by any NRC guidance documents or the IEEE standards for electrical equipment located in a mild environment. In summary the rigid application of Appendix QR-B to all active mechanical equipment is inconsistent with, and would be an unwarranted expansion of, existing regulatory guidance for the environmental design of such equipment. Its application would likely require the expenditure of significant additional licensee resources to address the more restrictive methods and documentation provisions of this nonmandatory Appendix. <u>NUGEQ Recommendation:</u> As suggested in Comment 2 the NRC should limit this revision of Regulatory Guide 1.100 to seismic and dynamic qualification and issued separate guidance on functional qualification of active mechanical equipment. Any regulatory positions that establish NRC expectations for complying with QME-1 Appendix QR-B should be deleted.	
WEC-1	B.1	*"Specifically, Sections B. 1 and C. 1 of this regulatory guide endorse, with exceptions and clarifications, the entire IEEE Std 344-2004 and Section QR "General Requirements," and Nonmandatory Appendix QR-A, "Seismic Qualification of Active Mechanical Equipment," of ASME QME-1 -2007 for the seismic qualification of electrical and active mechanical equipment, respectively."* <u>Comment (Editorial)</u> The word "respectively" should be deleted since there are more than two documents and all of the documents	The staff revised the statement for clarification.

		can be used in the seismic qualification of active mechanical equipment. Recommended Change Delete the word "respectively."	
WEC-2	B.1	*"Some solid-state relays and microprocessor-based components are quite fragile in terms of withstanding earthquake excitations."* Comment The following statement in our opinion has not been the case. "Some solid-state relays and microprocessor-based components are quite fragile in terms of withstanding earthquake excitations." We are not aware of any seismic issues that involve solid state relays. There are no solid-state relays and microprocessor-based components which we would consider fragile. The concern with microprocessors may be related to the connections to the buses and interfaces. Recommended Change The statement on solid state relays and microprocessors being sensitive should be deleted.	See IEEE-13
WEC-3	B.1	*"Recent studies related to the early site permit applications at certain hard-rock based plants along the east coast of the United States indicated that the site-specific spectra may exceed the certified design spectra of those new plants in the high-frequency range (20 hertz (Hz) and above)."* Comment DG-1175 defines high-frequency range as 20 Hz and above. It is understandable that an upper bound was not defined because it is dependent on the cutoff frequency of the hard rock site. The NRC should add a statement in this section to clarify. Recommended Change Further clarification should be added on how the upper limit to the high-frequency range should be defined.	See IEEE-8

WEC-4	B.1	*"Therefore, any attempt to use such past test experience data for the seismic qualification of high-frequency-sensitive equipment or fragile components in such plants clearly is not appropriate."* Comment This section excludes the use of previous seismic testing to address qualification of for high frequency sensitive equipment or fragile components because the high frequency motions were not intentionally input to the test. DG-1175 Section C. 1.1.1 .h specifies how new seismic qualification tests planned for equipment in plants with the high-frequency ground motion concern should be addressed. The criteria specified are already in IEEE Std 344-2004. Therefore, seismic test programs in compliance with IEEE Std 344-2004 (including seismic test motion) which have sufficient frequency content in the high-frequency range demonstrated through power spectral density (PSD) analysis should be acceptable. It is unclear why does DG-1 175 call out "fragile components" and what is the definition? Recommended Change Update section to allow pass seismic test data to permitted for addressing high frequency conditions as provided the data is in compliance with IEEE Std 344-2004 and demonstrates sufficient frequency content in the high-frequency range. Provide addition information as to the definition and usage of the term "fragile components."	See IEEE-5
WEC-5	B.2	Comment DG-1175 Section B.2 provides information associated with functional qualification of active mechanical equipment. Regulatory Guide (RG) 1.148 also provides information on functional specification of active valves and primarily endorses ANSI N278.1-1975. Functional qualification of active mechanical equipment discuss in DG-1 175 may be better suited for RG 1.148 since it presently exists. Recommended Change	See IEEE-10

		Recommend that functional qualification of active mechanical components not related to seismic qualification be discussed in a revision to RG 1.148. RG 1.100 should only provide guidance in the area of seismic qualification of electric and mechanical equipment. DG-1 175 Section B.2 (Functional Qualification of Active Mechanical Equipment) should be removed and the title of DG-1 175 should revert back to "Seismic Qualification of Electrical and Mechanical Equipment for Nuclear Power Plants."	
WEC-6	C.1.1.1c	*"(2) fragile electronic components, such as solid-state relays and microprocessors-based components;..."* <u>Comment</u> The phrase "fragile electronic components" in our opinion has not been observed in the seismic qualification of solid-state relays and microprocessor-based components. There are no solid-state relays and microprocessor-based components which we would consider fragile. The concern with microprocessors may be related to the connections to the buses and interfaces. <u>Recommended Change</u> The condition "(2) fragile electronic components, such as solid-state relays and microprocessors-based components: should be deleted.	See IEEE-13
WEC-7	C.1.1.1g	*"Furthermore, credit should not be taken for the inadvertent high frequencies present in some of the IEEE-344-type seismic qualification tests of equipment in the past, which may have shown the ZPA of the TRS to be up to 100 Hz."* <u>Comment</u> Request further clarification as to why this position is taken in DG-1 175. As written the statement would exclude the use of previous testing to address high frequency concerns since the test motion did not intentionally require input in the high frequency range. If an evaluation of the test input is performed and the data	See IEEE-17

		demonstrate sufficient frequency content in the high-frequency range throughout the time history through PSD analysis then the data should be acceptable. This approach is consistent with regulatory guidance in Section C. 1.1.1 h (also Section C.1.2.1h). We believe IEEE Std 344-2004 provides sufficient guidance to ensure that the input is generated and in compliance with the frequency range of interest. IEEE Std 344-2004 Annex B defines how to verify the test data has sufficient content over the frequency range of interest throughout the input time history. Recommended Change Clarify that the subject test data is not acceptable unless further evaluation is performed and data generated to demonstrate there is sufficient frequency content over the frequency range of interest.	
WEC-8	C.1.1.1i	*"Electric equipment should be qualified with five one-half SSE events followed by one full SSE event (SECY-93-087) (Ref. 28) even if the OBE of a plant is defined to be one-third of SSE or less. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events may be used in accordance with Annex D, "Test Duration and Number of Cycles," to IEEE Std 344-2004, when followed by one full SSE."* Comment The DG-1175 position does not recognize that some plants are licensed with an OBE that is greater or less than one-half SSE. The document SECY-93-087 addressed issues affecting Advanced Light-Water Reactors (ALWRs), for which the OBE eliminated from design certification when the OBE is established at less than or equal to one-third the SSE. It also states the following: *"With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE Standard 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE*	See IEEE-18

		event (with 10 maximum stress cycles per event). Of these alternatives, the staff concludes that equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE." Recommended Change This section should be updated to identify the present wording is associated with qualification of equipment for new plant designs. Wording should also be added to identify for other applications the OBE requirement is based on plant specific licensing requirements. [For Section C.1.2.1j, Page 14 the recommended change is applicable to active mechanical equipment.]	
WEC-9	C1.1.1j	*"The IEEE Std 344-2004 recommended no damping values. The damping values listed in Table 6 of NRC Regulatory Guide 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," (Ref. 29) issued in March 2007, are recommended. These damping values are the updated values currently acceptable to the NRC staff."* Comment DG-1175 is recommending use of NRC Regulatory Guide 1.61, Revision 1 damping values. This is not appropriate since older plants as well as AP1000 uses damping values consistent with Regulatory Guide 1.61, Rev. 0. In addition, IEEE Std 344-2004 sub-clause 6.3.1 (Application of damping in analysis) identifies "Appropriate values of damping may be obtained from tests or other justifiable sources." IEEE Std 344-2004 sub-clause 6.3.2 (Application of damping in testing) and 8.6.1.3 (Damping selection) identify for testing "The RRS are usually specified at several levels of damping. When available, the RRS with a damping of 5% is the recommended choice for use in testing." Recommended Change	See IEEE-19

		This section should be reworded to indicate the version of Regulatory Guide 1.61 as included in the plant licensing basis. This sentence dealing with damping in IEEE Std 344-2004 should also be deleted.	
WEC-10	C.1.1.2k	"A coherence function of less than 0.5 and an absolute value of the correlation coefficient function of less than 0.3 are not acceptable. The NRC positions on the numerical values for the coherence function and the correlation coefficient function for defining statistically independent motions are the same as in Reference 34, particularly the following: **i.** For the coherence function, numerical values ranging from 0.0 to a maximum of 0.3 and an average of approximately 0.2 are acceptable. ii. An absolute value of less than 0.16 for the correlation coefficient function is acceptable." <u>Comment</u> The coherence function and correlation coefficient limits appear to be restrictive. IEEE Std 344-2004 and IEEE Std 344-1987 specifies that either coherence function and correlation coefficient limits criteria must be met for the shake table test to be valid. That is: either the coherence function must be less than or equal to 0.5 at all frequencies of interest or the correlation coefficient need be less than 0.3. Both criteria need not be passed, just one or the other. The coherence function and cross correlation coefficient were originally developed in ASME Paper 83-PVP-22 based on his review of several actual earthquakes and used in the development of requirements initially in IEEE Std 344-1987. We are not aware of any new industry data which would change this position. In addition, Reference 34 (ASME Boiler and Pressure Vessel Code, Section III Division 1, Article N-1213.1 of Nonmandatory Appendix N) of DG-1175 is addressing the development of time history input for analysis where you are developing inputs associated	See IEEE-25

		with a specific in-structure required response spectrum. Where as, for seismic testing, the inputs are normal generic in nature (multiple plant sites/locations) and the RRS will be most likely the same in both horizontal axes as a minimum. <u>Recommended Change</u> This section should be updated to concur with the present criteria in IEEE Std 344-2004 for test input generation associated with coherence function and correlation coefficient limits and its usage.	
WEC-11	C.1.2.1a	*"In endorsing the use of ASME QME-1-2007, the staff noticed that several appendices are designated as either nonmandatory or mandatory (e.g., Nonmandatory Appendix QR-A; Nonmandatory Appendix QR-B; Nonmandatory Appendices QDR-A, QDR-B, and QDR-C; Nonmandatory Appendices QP-A, QPB, QP-C, QP-D, and QP-E; and Mandatory Appendix QV-1). The staff position is that, once the user commits to the use of ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, the criteria and procedures delineated in those appendices then become the requirements for its qualification program, unless the deviations are justified."* <u>Comment</u> RG 1.148 may be a more correct place for the Operability portion of QME-1. Including the operability portions of ASME QME-1-2007 into DG-1175 may create a potential conflict with RG 1.148. DG-1175 indicates that 'The staff position is that, once the user commits to the use of ASME QME-1-2007 for its qualification of active mechanical equipment in NPPs, the criteria and procedures delineated in those appendices then become the requirements for its qualification program, unless the deviations are justified." ASME QME-1-2007 includes Nonmandatory Appendix QV-A "Functional Specification for Active Valves for Nuclear Power Plants." This nonmandatory appendix	See IEEE-10

		represents a potential for conflict with RG 1.148. RG 1.148 Value/Impact Statement, Section Value (page 5) states, "It is anticipated that the most important contributions from ANSI N278.1-1975 will be realized when subsequent standards, which are currently being developed to address such topics as valve assembly functional qualification and production, are in place to provide a set of requirements covering various aspects of valve assembly operability." ASME QME-1-2007 represents the latest development in valve assembly functional qualification and production indicated. While it is not specifically noted that RG 1.148 will be revised to endorse these requirements it seems logical that all requirements regards functional qualification should be gathered into a single regulatory position. Because RG 1.148 already addresses some portion of functional qualification it would be the logical place for all functional qualification to be gathered. RG 1.100 has previously only addressed seismic qualification which is only of functional qualification. <u>Recommended Change</u> Recommend regulations dealing with ASME QME-1-2007 in the area functional qualification be moved to RG 1.148.	
WEC-12	C.1.2.1g	"*For certain hard-rock-based plants, the site-specific spectra may exceed the certified design spectra in the high-frequency range. This guide refers to this phenomenon as the high-frequency ground motion concern. As a result of the high-frequency ground motion, the seismic input to SSCs may also contain high frequency excitations. For operating BWR plants, the seismic qualification of some safety-related active mechanical equipment were performed using IEEE-344-type tests with intentional high-frequency contents to account for concurrent BWR hydrodynamic loads. However, the vast majority of existing seismic qualification tests used input frequencies up to only 33 Hz. These past test experience data are therefore not*	See IEEE-17

		acceptable for the seismic qualification of high frequency- sensitive equipment or fragile components. Furthermore, credit should not be taken for the inadvertent high frequencies present in some of the IEEE-344-type seismic qualification tests of equipment in the past, which may have shown the ZPA of the TRS to be up to 100 Hz. Ball joints and kinematics linkages of the shake tables could have generated these inadvertent high frequencies, and the NRC staff considers them to be noise signals that may not have the proper frequency content with sufficient energy to be compatible with the amplified region of the RRS at high frequencies." <u>Comment</u> Request further clarification as to why DG-1175 (Regulatory Positions on ASME QME-1) discusses high frequency response. The DG-1175 should limit discussions and positions to high frequency sensitive equipment. We believe that mechanical equipment is not sensitive to high frequency. DG-1175 position on high frequency sensitive equipment should only be applied to sensitive electrical component which may be attached to the mechanical equipment. As written the statement would exclude the use of previous testing to address high frequency concerns since the test motion did not intentionally require input in the high frequency range. If an evaluation of the test input is performed and the data demonstrate sufficient frequency content in the high-frequency range throughout the time history then the data should be acceptable. This approach is consistent with regulatory guidance in Section C. 1.1.1.h. We believe IEEE Std 344-2004 provides sufficient guidance to ensure that the input is generated and in compliance with the frequency range of interest. IEEE Std 344-2004 Annex B defines how to verify the test data has sufficient content over the frequency range of interest throughout the input time history. Therefore,	

		seismic test programs in compliance with IEEE Std 344-2004 (including seismic test motion) which have sufficient frequency content in the high-frequency range demonstrated through PSD analysis should be acceptable. <u>Recommended Change</u> Update section to clarify that electrical component which may be attached to the mechanical equipment may be high frequency sensitive and are address by this section. Allow pass seismic test data to permitted for addressing high frequency conditions as provided the data is in compliance with IEEE Std 344-2004 and demonstrates sufficient frequency content in the high-frequency range.	
WEC-13	C.2.1	"*In general, the NRC staff finds ASME QME-1-2007 acceptable for the functional qualification of* ***(1)*** *active mechanical equipment in new NPPs; and (2) new addition or replacement of active mechanical equipment in operating NPPs, subject to the following provisions:*" <u>Comment</u> Section C.2 of DG-1175 provides information associated with functional qualification of active mechanical equipment. Regulatory Guide (RG) 1.148 also provides information on functional specification of active valves and primarily endorses ANSI N278.1-1975. Functional qualification of active mechanical equipment discuss in DG-1175 may be better suited for RG 1.148 since it presently exists. <u>Recommended Change</u> Recommend that functional qualification of active mechanical components not related to seismic qualification be discussed in a revision to RG 1.148. RG 1.100 should only provide guidance in the area of seismic qualification of electric and mechanical equipment. DG-1 175 Section B.2 (Functional Qualification of Active Mechanical Equipment) should be removed and the title of DG-1 175 should revert back to "Seismic Qualification of Electrical and Mechanical	See IEEE-10

		Equipment for Nuclear Power Plants."	
Dom-1	General	The draft RG title and scope have been changed to include functional qualification of active mechanical equipment, as compared to the two previous revisions of RG 1.100 which only discussed seismic qualification of electrical and mechanical equipment. This change is because the RG now endorses ASME QME-1-1994, which covers functional qualification of active mechanical equipment. The main discussion on pages 5 through 8 of the DG is for active, motor-operated valves. It is noted that RG 1.148 also discusses functional specification of active valves and primarily endorses ANSI N278.1-1975. Although the ANSI standard by itself does not provided complete assurance of operability, there is an overlap between DG-1175 and RG 1.148 for functional qualification of active valves. It is recommended that functional qualification of active mechanical components (which have no direct bearing on seismic qualification) should be discussed in a revision to RG 1.148. RG 1.100 should provide guidance just for seismic qualification of electric and mechanical equipment. Either remove functional qualification of active mechanical equipment from this DG or reconcile the overlap between DG-1175 and RG 1.148 in another manner.	See IEEE-10
Dom-2	B.1 C.1.1.1b C.1.1.2b,c	In the SERs that NRC sent to the USI A-46 plants in the past, it was stated that older vintage plants could use the experience-based SQUG-GIP method for seismic verification of new and replacement equipment provided they revised their licensing bases. Many older plants are currently using the SQUG-GIP method. The DG is silent on this. Add a sentence at the end of this paragraph to this	See NEI - 2

		effect: “However, older vintage plants can, with a few exceptions, use the experience-based SQUG-GIP method for seismic verification of new and replacement equipment provided they revise their licensing bases via safety evaluations”. Alternatively, reconcile the fact in the DG that NRC has previously accepted earthquake experience-based qualification of new/replacement equipment in older plants.	
Dom-3	B.1 C.1.1.1g C.1.2.1g	The high frequency content, which exists in most existing tests, whether inadvertent or deliberate, will still be imparted to an item on equipment on the shake table. Therefore, high frequency vibratory motions generated on a shake table in an inadvertent manner may not be inconsequential. The DG should clarify that such inadvertent motions can be credited provided they are shown to meet stationarity requirements per Appendix B of IEEE Std 344-1987 or 2004 (when one of these versions of the IEEE Standard is the plant’s commitment). However, in IEEE Std 344-1975, there was no requirement for stationarity check. For example, previous seismic shake tests for BWR Mark II and III plants (committed to the 1975 version of the standard) were frequently utilized to qualify equipment for the combined seismic and hydrodynamic loads with high frequency content up to 100 Hz and were accepted by the NRC staff in SQRT audits. Revise this section appropriately, such as adding a sentence to this effect: “When the existing seismic tests contain inadvertent high frequency motions due to ball joints and kinematics linkages, such tests shall be shown to meet the stationarity requirements discussed in Appendix B of IEEE Std. 344-2004.”	See IEEE-5 and IEEE-17
Dom-4	B.1 C.1.1.1.i C.1.2.1.j	In the last sentence of this paragraph, it says that the test sample shall be subjected to simulated OBE and SSE vibrations per IEEE Std. 344-2004. In section	See IEEE-18

		C.1.1.1i (p. 10) two alternatives for the number of tests/cyclic considerations are provided. However, another alternative when OBE is defined as 1/3 or less of SSE is to use two SSE events with 10 maximum stress cycles per event in accordance with SRP 3.7.3 (p. 4), March 2007. This alternative should also be listed. Revise these sections to include an option that 2 SSE tests, as an alternative to 5 OBE and 1 SSE are also acceptable when the OBE is designated as 1/3 or less of the SSE.	
Dom-5	C.1.1.1.j	The IEEE Std. 344-2004 has a section on damping. While the damping values in RG 1.61 can be used when qualification is by analysis, there should be no specific requirement on damping values to be used for shake-testing, only that the equipment damping at which the RRS is developed should be the same or lower than the TRS damping value. Clarify the statement in this section that for qualification by shake-table testing, RRS with any reasonable damping value (such as 5% of critical damping) can be used provided that the TRS is also plotted at the same damping value or a higher damping value.	See IEEE-19
Dom-6	General	There is no discussion of required margins for seismic testing, except in Section C1.1.2d re. test experience spectra. A 10% margin is recommended in IEEE Std 323. Also, SRM on SECY-93-087 states that the Commission approved the use of a 1.67 margin over SSE for a margin type assessment. The intent of these margins should be clarified, particularly for seismic testing. The required margins and/or the intent of margins in TRS vs. RRS over the applicable frequency range should be discussed in the RG so that there is no	See IEEE-22

		confusion by the practitioners.	
ASME-1	General	The mechanical equipment functional qualifications included in DG-1175 are an expansion of RG-1.100 and overlaps with several older NRC documents (Regulatory Guides and Standard Review Plan). There needs to be a discussion to foster a better understanding of the regulatory position with regard to mechanical equipment functional qualification and QME-1-2007 requirements.	Also see NUGEQ -2 and NUGEQ-3.
ASME-2	General	The restrictions on the use of experience-based seismic qualification to USI A-46 power plants results in this method of seismic qualification being disallowed by the DG for new plants. The experience-based seismic methods have been in developed and used by the nuclear industry for quite some time. These methods were approved by the consensus committee process based on sound and accepted engineering judgment, information, and practices, and ASME requests that use of experience-based methods be allowed and accepted.	As delineated in C.1.1.1b, the use of experience-based method for seismic qualification of electric equipment will be subject to the review and approval by the NRC staff. Even though IEEE Std 344-2004 and ASME QME-1-2007 indicated limitation of earthquake or test experience-based qualification, the staff found that there are difficulties to justify the demonstration of similarity in seismic excitation, physical , functional, and dynamic characteristics between electric equipment in the experience database and those in the NPP to be seismically qualified.
ASME-3	General	ASME and IEEE need to work together in order to better define scope and responsibility of each of our respective organizations. For example, we should cross-reference requirements between each of our standards rather than to duplicate them. Redundant standards documents cause confusion and may make it very difficult for NRC to provide regulatory endorsement and appropriate guidance on their application.	The NRC staff will continue to work with IEEE and ASME in developing standards documents.
ASME-4	General	The NRC has made the QME Nonmandatory Appendices mandatory. The intent of the QME standard is to provide an acceptable method to meet a particular qualification requirement while providing some flexibility for a user. If the Nonmandatory Appendix is committed to by a user, all aspects of that Nonmandatory Appendix	The staff agreed that Mandatory Appendices contained provisions must be followed. Compliance of Nonmandatory Appendices, which provided information or guidance, is voluntary.

		become mandatory. There needs to be a better understanding of what the minimum requirements are and when it is appropriate to have non-mandatory approaches for equipment qualification.	However, if a user commits to use any Nonmandatory Appendices for its qualification of active mechanical equipment in NPPs, all the criteria and procedures that delineated in both the Mandatory Appendices and those committed Nonmandatory Appendices then become the requirements for its qualification program. Justification must be provided for any deviations, which will be subjected to NRC staff review and approval.
Duke Energy-1	General	Duke supports and adopts the comments submitted by the Institute of Electrical and Electronics Engineer (IEEE) Nuclear Power Engineering Committee and the Nuclear Energy Institute by letters dated July 10, 2008 and July 11, 2008, respectively	The staff has reviewed and provided responses to the comments from IEEE NPEC committee and Nuclear Energy Institute.

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Regulatory Applications of Computer Codes

ACRS Meeting July 8, 2009
Ralph R. Landry
Senior Level Advisor
NRO/DSRA

Purpose of Code Application

- Confirmation of Submittals
  – Are licensee/applicant analyses reasonable?
  – Have submittals captured the phenomena?
- Exploratory
  – Are there any hidden “cliffs” that have not been discovered?
- Resolution of Generic Issues

NRO Use of TRACE

- Code Applicability Report for Each Design
  - Assess for unique features
  - Does the plant model perform reasonably
- Bounding Calculations for Comparison
  - Applicants use of parametric sampling
  - Do we see the trends and phenomena?

NRO Use of TRACE

- Input Model for Each New Reactor Design
- High Comfort Level with TRACE

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Introduction to TRACE Applicability to ESBWR LOCA

ACRS Meeting 7/8/2009

Joseph Staudenmeier
RES/DSA/CDB

ESBWR Design Features

- Classical BWR ECCS System Strategy
  - Isolate reactor on leak indication
  - Activate ADS on low level trip
  - Depressurize to low pressure injection
- No large liquid breaks as in jet pump plants
- No fuel cladding heatup during design basis LOCAs

ESBWR Unique Design Features

- Gravity Driven Cooling System
- Passive Containment Cooling System

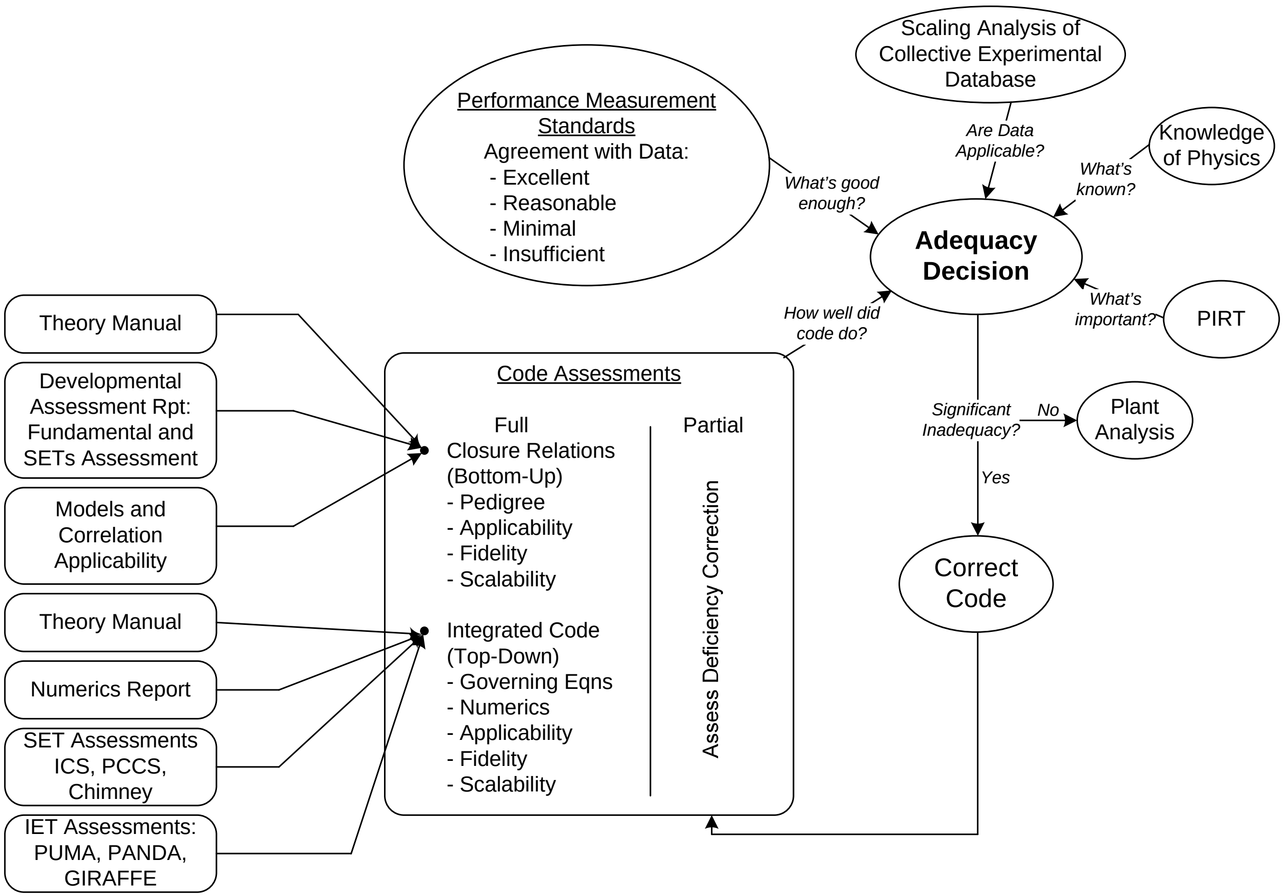
ESBWR ECCS Research Program

- TRACE Model Development
- TRACE Assessment
- Report on TRACE Adequacy for ESBWR LOCAs
- PUMA-E Confirmatory Testing

TRACE Adequacy for ESBWR

- The capabilities and limitations of **TRACE Version 5.0** were evaluated for predicting the phenomena important for ESBWR loss-of-coolant accidents (LOCAs)
- Evaluation method and results documented in “Adequacy of TRACE Version 5.0 for Simulating ESBWR Loss-of-Coolant Accidents,” Information Systems Laboratories, Inc., ISL-NSAD-TR-08-01, October 2008 (Proprietary):
  - ESBWR LOCA Phenomena Identification and Ranking Table (PIRT) evaluation
  - TRACE code documentation reviews
  - Development of consistent modeling approach for ESBWR plant and test facilities used for code assessment
  - TRACE code assessments against experimental data
  - User guidelines and cautions for TRACE ESBWR modeling

Adequacy Demonstration Process



TRACE Model Development for ESBWR

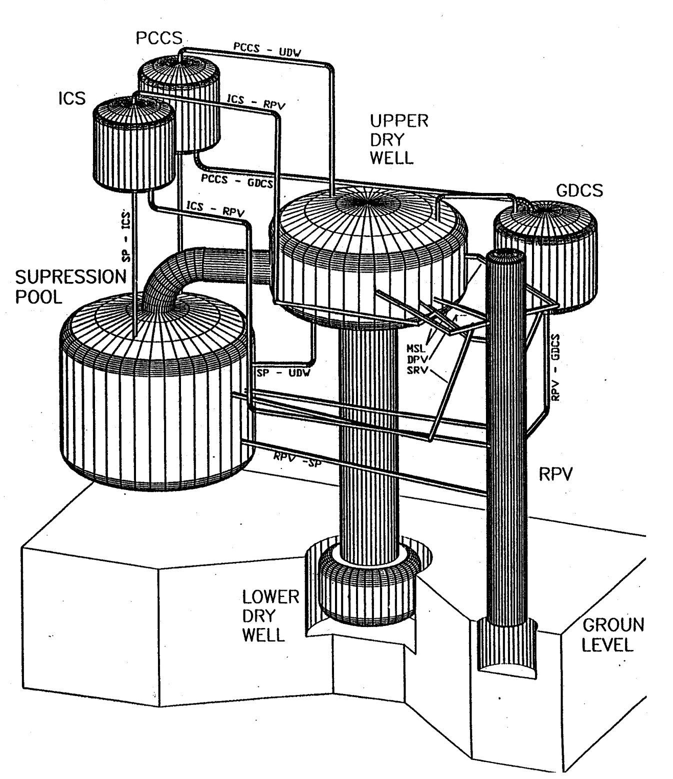
- Film Condensation Model
  - Treats pure steam and mixtures of noncondensable gas and steam
  - Applicable to PCCS, ICS, containment walls

TRACE Assessment for ESBWR

- Assessment of thermal-hydraulic phenomenon common to all reactors
  - void fraction, heat transfer, etc.
  - TRACE 5.0 Assessment Manual Appendix B: Separate Effects Tests in ADAMS ML071200466
- Film condensation component tests for new model
- Integral tests applicable to ESBWR

PUMA-E Integral Testing for ESBWR

- Confirmatory testing was performed at the PUMA facility.
  - Study system behavior and interactions of the ESBWR safety systems
  - Provide additional data for code assessment
- Behavior of the PUMA-E tests are understood and are qualitatively the same as the PUMA SBWR tests



ESBWR ECCS Calculations

- Calculations were performed for a range of break locations and sizes
  - Main Steam Line Break
  - Feedwater Line Break
  - Isolation Condenser Line Break
  - Gravity Driven Cooling System Break
  - Bottom Drain Line Break
- Sensitivity calculations were performed to examine the effect of model deficiencies uncertainties
- The calculated response of the ESBWR ECCS is predictable and consistent with integral test results

ESBWR Research Conclusions

- The calculated performance and response of the ESBWR ECCS is predictable and consistent with integral test results
- TRACE is adequate as an audit tool for analyzing the ESBWR ECCS system response